Project No. 12-4733

Integral Inherently Safe Light Water Reactor (I^2S-LWR)

Integrated Research Project

Bojan Petrovic
Georgia Institute of Technology

In Collaboration with:
Brigham Young University
Florida Institute of Technology
Morehouse College
University of Idaho
University of Michigan
University of Tennessee
Virginia Polytechnic Institute and State University

Damian Peko, Federal POC
Don Williams, Technical POC

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Integral Inherently Safe Light Water Reactor (I^2S-LWR)

Final Report

Prepared by:
Bojan Petrovic (PI)
On behalf of the ^{2}S-LWR Project Team (please see the inner cover page)

Submitted by:
Bojan Petrovic (Bojan.Petrovic@gatech.edu), Project PI
Nuclear and Radiological Engineering
Georgia Institute of Technology
770 State St., Atlanta, GA 30332-0745

Performed Under:
NEUP 12-4733, SRC#00132015
Under Prime Contract No. DE-AC07-05ID14517
(GT Project 2506J12, Research Agreement RD537)

POC: Don Williams
Federal Manager: Damian
Peko

Rev. 0
December 2016

I²S-LWR Project Team

Lead organization:
Georgia Institute of Technology, Atlanta, Georgia
Bojan Petrovic (PI), Farzad Rahnema (Co-PI)
Chaitanya Deo, Srinivas Garimella, Preet Singh

Team organizations:

Academia:
Brigham Young University
Matthew Memmott (Co-PI)

Florida Institute of Technology, Florida
Guy Boy (Co-PI)

University of Idaho, Moscow, Idaho
Indrajit Charit (Co-PI)

University of Michigan, Ann Arbor, Michigan
Annalisa Manera (Co-PI), Thomas Downar, John Lee

Morehouse College, Atlanta, Georgia
Lycurgus Muldrow (Co-PI)

University of Tennessee, Knoxville, Tennessee
Belle Upadhyaya (Co-PI), Wesley Hines

Virginia Tech, Arlington, Virginia
Ali Haghighat (Co-PI)

Industry:
Westinghouse Electric Co., Cranberry Township, Pennsylvania
Paolo Ferroni (Co-PI), Fausto Franceschini, David Salazar, William Mack,
Jason Young, Alex Harkness, Robert Ammerman, Matthew Smith

Southern Nuclear Operating Company, Birmingham, Alabama – Nick Irvin

National Laboratory Collaboration:
Idaho National Laboratory (INL), Idaho Falls, Idaho
Abderrafi M. Ougouag (Co-PI), George Griffith

International Research Collaboration:
University of Cambridge, Cambridge, United Kingdom
Geoffrey Parks (Co-PI)

Politecnico di Milano, Milano, Italy
Marco Ricotti (Co-PI)

University of Zagreb, Zagreb, Croatia
Nikola Čavlina, Davor Grgić, Dubravko Pevec (Co-PIs)

Consultant:
Hans Garkisch

Acknowledgments

This research was performed using funding received from the DOE Office of Nuclear Energy's Nuclear Energy University Programs (NEUP), Integrated Research Project (IRP) 12-4733 “Integral Inherently Safe Light Water Reactor (I^{2}S-LWR)”

This multidisciplinary research project was performed with contributions from many individuals working on different project areas, including the one covered in this Topical Report. Therefore, it was not practical to list everybody that was in some way involved on the cover page of each Topical Report. Instead, the main Final Report provides a unified list of all contributors over the whole course of the project. The cover page of each individual Topical Report lists only the authors and main contributors to that report.

Continuous support and guidance of the Federal Program Manager, Mr. Damian Peko, and Technical Reviewer, Mr. Donald Williams, Jr., contributed to the quality and outcomes of the project and are highly appreciated.

Disclaimer

(This page intentionally left blank)

Table of Contents

List of Figures

List of Tables

1. Executive Summary

This final report summarizes results of the multi-year effort performed during the period 2/2013-12/2016 under the DOE NEUP IRP Project "Integral Inherently Safe Light Water Reactors (I²S-LWR)". The goal of the project was to develop a concept of a 1 GWe PWR with integral configuration and inherent safety features, at the same time accounting for lessons learned from the Fukushima accident, and keeping in mind the economic viability of the new concept. Essentially (see Figure 1-1) the project aimed to implement attractive safety features, typically found only in SMRs, to a larger power (1 GWe) reactor, to address the preference of some utilities in the US power market for unit power level on the order of 1 GWe.

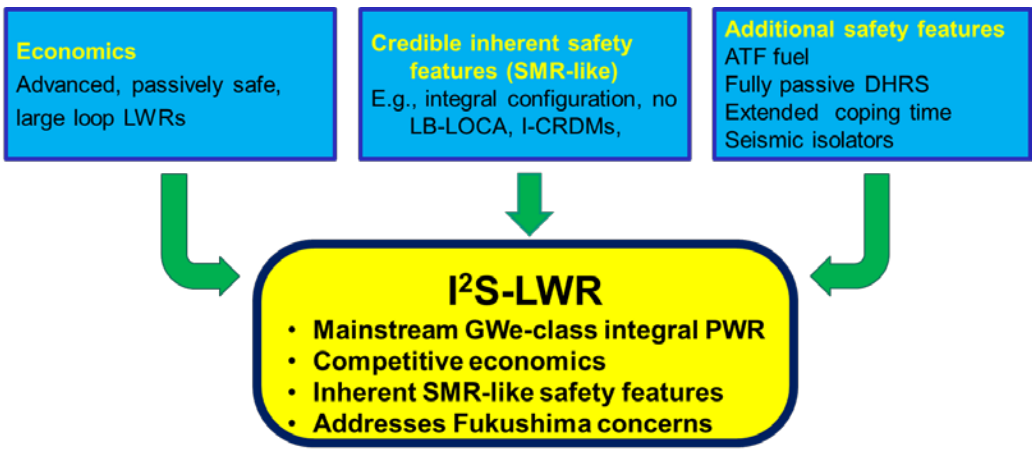


Figure 1-1. I²S-LWR concept

The project was performed by a multi-disciplinary multi-organization team of 14 organizations, lead by Georgia Tech and including seven other US universities (Brigham Young University, Florida Institute of Technology, University of Idaho, University of Michigan, Morehouse College, University of Tennessee, and Virginia Tech), nuclear industry and utility (Westinghouse Electric Co. LLC and Southern Nuclear), national laboratory (Idaho National Laboratory), and three international academia partners (University of Cambridge, UK; Politecnico di Milano, Italy; and, University of Zagreb, Croatia). This diverse expert team ensured successful completion of the project, while the participation of industry provided a valuable practical expertise and sanity-check throughout the course of the project.

In addition to about 30 Co-PIs and senior team members, the project engaged 10 young faculty, researchers, scientists and post-docs, as well as close to 30 graduate (MS and PhD) students, and over 70 undergraduate students, most of them through senior design projects. Thus, more than a hundred young faculty/researchers and students were trained and had opportunity to work on a cutting-edge research, under realistic real-life R&D conditions. This education and training by itself provides an excellent "return on investment" to DOE.

The External Advisory Board (EAB) was established at the very beginning of the project. The EAB consisted of experienced senior executives: Regis Matzie (Panel Chair), retired Chief Technology Officer

(CTO) of Westinghouse Electric Co.; John McGaha, retired Senior Executive of Entergy Nuclear; Ted Marston, retired CTO of EPRI and Principal of Marston Consulting; Albert Machiels, Senior Technical Executive, Electric Power Research Institute; Lynn Weaver, retired President of Florida Institute of Technology; and Chuck Kling, retired Consulting Engineering of Westinghouse Electric Co. The EAB met twice a year and provided valuable feedback to the team.

Results have been documented in 14 quarterly reports, some multi-volume, and uploaded to PICS, totaling over 3,300 pages. Therefore, it is impossible (and would be counterproductive) repeating all the details in this final report; instead, it aims to capture main findings and results. The reports have been carefully reviewed by the DOE Technical Monitor, Don Williams (ORNL) and Federal Program Manager, Damian Peko. Their questions and comments contributed to maintaining the quality and resolving issues that appeared during the course of the project. Regular annual briefings on the project status were given to the DOE leadership. Additionally, the projects was presented at several NRT and AFC integration meetings, where a strong synergy with the DOE ATF efforts was observed.

Results have been disseminated through journal articles and conference papers (over 60 total), as well as presentations at conferences and technical seminars. Additionally, presentation of the project to a broader audience has been achieved through: (a) organizing a special panel session at the 2014 ANS Winter meeting (presenting the initial results); and (b) organizing and editing two volumes of a special issue of Annals of Nuclear Energy devoted exclusively to I²S-LWR to capture main results and findings. (Volume 1 was published in 2/2017; Volume 2 expected by end 2017.)

The project was guided by the top level requirements (Table 1-1) that were established in the proposal, and updated and somewhat expanded during the early project phase. The requirements were formulated in terms of hard 'must satisfy' requirements, with additional soft 'it would be valuable to satisfy' stretch targets. All hard requirements have been met; in addition, some of the stretch targets have been met as well. Last column in Table 1-1 summarizes how the final design met the requirements.

Over the first project year, trade-off studies were performed and down-selections made, followed by establishing the preliminary concept, fairly detailed for the stage of the project, but still lacking design information in certain areas. During the second project year, most of the missing areas and details were covered resulting in a modified concept. Critical review over the third project year led to further improvements and harmonization, resulting in the final concept. Potential path forward was also addressed at the end of the project.

Table 1-1: Top level requirements.

	Requirement	Stretch Target	Comment	Final
APPLICATION-DRIVEN REQUIREMENTS				
Power	~1,000 MWe		For markets preferring large plants; economy of scale	985 MWe
Electricity production efficiency	>32%	35%	Economic competitiveness; reduced reject heat	>34%
Design lifetime	60 years	100 years	Competitiveness; economics, sustainability	100 years
Reactor pressure vessel	Same size as or smaller than current large PWRs (e.g., EPR, ABWR) [~5m]		Manufacturability	490 CM I.D.
FUEL-RELATED REQUIREMENTS				
Fuel/cladding system	Enhanced accident tolerance		Post-Fukushima considerations	Clad: reduced oxidation rate Fuel: high conductivity
Fuel enrichment	Viable reloading with <5% enriched fuel	Potentially improved fuel cycle with 5-8% enriched fuel	Possibility to use existing infrastructure for <5% enrichment Stretch target abandoned due to reduced industry interest for >5% enriched fuel	Licensed <5% enriched fuel
Refueling	Multi-batch, refueling interval 12 months or longer	Options for 12-18-24 months refueling	24-mo cycle when required by utilities 24-mo cycle possible but may require >5% enr for competitive FCC (reduced industry interest)	12-month and 18-month refueling scheme developed
SAFETY AND SECURITY				
Security, safeguards and proliferation resistance	As in current LWRs or better		Compact partly under grade nuclear island	Low profile; improved physical protection
Safety indicators	CDF <3x10^{-7} LERF <3x10^{-8}	CDF <1x10^{-7} LERF <1x10^{-8}	Improve safety indicators relative to current Gen-III+ passive plants	Preliminary PRA meets stretch target
Safety philosophy/systems	Inherent safety features Full passive safety High level of passivity		Eliminate accident initiators. Eliminate need for offsite power in accidents	Full passive safety. LB-LOCA and rod ejection eliminated.
Grace period	At least 1-week	Indefinite for high percentage of considered	Resistance to LOOP and Fukushima-type scenarios	At least 1 week. Indefinite in many scenarios.

		scenarios, with no/minimum operator action		
Decay heat removal	Passive system with air as the ultimate heat sink		Resistance to LOOP and Fukushima-type scenarios	Passive 3-of-4 DHRS-to-air designed
Seismic design	Single compact building design	Seismic isolators	Allows siting at many locations	Compact footprint, Seismic isolators
Other natural events	Robust design		Address unforeseen events	Robust design
Monitoring	Enhanced, in normal and off-normal conditions		Improve normal operation; Address unforeseen events	Robust self-diag I&C algorithms
Spent fuel pool safety	Monitoring Passive cooling		Address Fukushima issues with SFP	Passive air-cooled DHRS
Used nuclear fuel management	On-site, for the life of the plant		Remove reliance on repository availability at certain date	10 year SFP pool and dry cask storage.
DEPLOYMENT REQUIREMENTS				
Economics	Competitive with current LWRs		Differential economics	Diff. econ. indicates competitive
Deployment	Near-term: feasible with <5% enriched oxide fuel	Long-term option: up to 8% enriched if industry interest	Oxide fuel provides path to accelerated deployment	Option to start with current fuel; mid-term FeCrAl and U3Si2; long-term SiC clad
Operational flexibility	2-batch and 3-batch, ≥12-month cycle	5% and 8% 12-18-24 months cycle	Diverse market needs Currently reduced interest for >5% enrichment fuel; therefore not developed	Focus on <5% fuel for now.
Operational flexibility		Load follow with MSHIM	Reduced effluents (environmental)	MSHIM assessed feasible but not developed in detail
D&D	Returned to green site simplified		Sustainability and public acceptance	Reduced activation outside RPV simplifies D&D and reduces dose

The main plant parameters are given in Table 1.2. More details are provided in the subsequent Chapters of this report. I²S-LWR reactor vessel is shown in Figure 1.2 (external view and cut-through view to show internals).

Table 1.2: I^2S-LWR main parameters.

PARAMETER	VALUE
Thermal Power	2850 MWth
Electric output (net)	~1000 MWe (986 MWe for the considered environmental temp.)
Net thermal efficiency	34.6%
Vessel inner clad I.D.	490 cm
Vessel height	22.87 m
Primary circuit	Integral configuration
Primary pressure	2250 psi
Core	121 fuel assemblies
Fuel assemblies	19x19 lattice (336 fuel rods, 24 control rod guides, 1 central instrumentation guide tube)
Active core height	12 ft
Enrichment	<5%
Refueling	18- and 12-month; 2- and 3-batch
Reactor coolant pumps	8
Primary heat exchangers (PHE)	Microchannel type (MCHX), 8 modules, paired in 4 subsystems
Steam generating system	PHE and flashing drums
Decay heat removal system	Fully passive, 4 trains, ambient air ultimate heat sink
Containment O.D.	23 m
Containment height	52 m
Plant design life	>100 years (based on reactor vessel fluence)

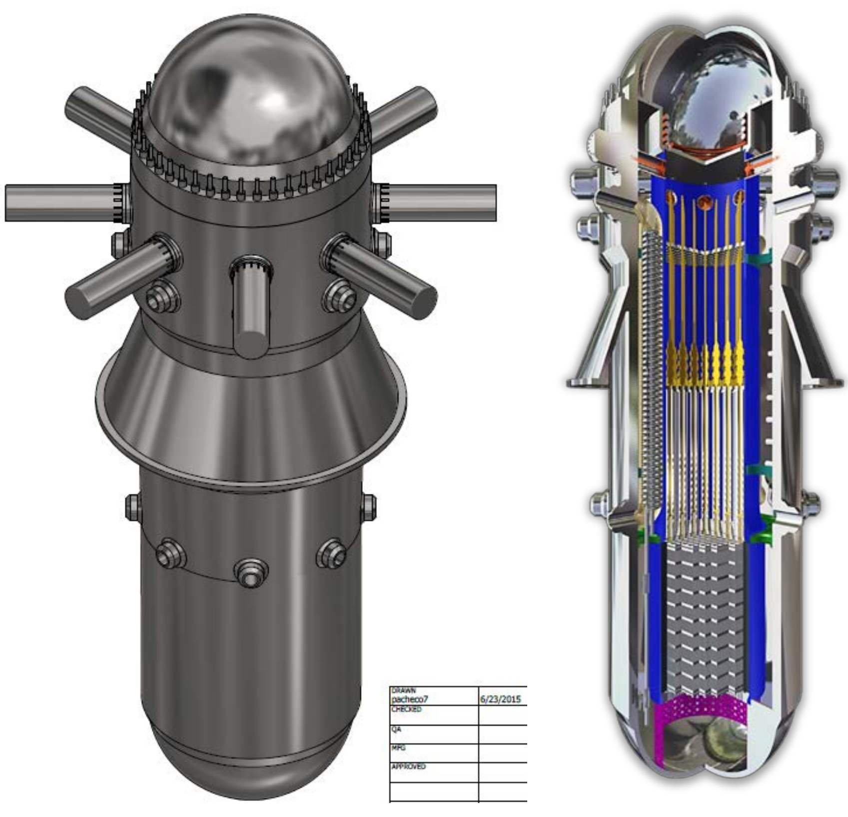


Figure 1.2. Reactor vessel – external view and cut-through

Nuclear Island building is shown in Figure 1.3, while the general site layout is presented in Figure 1.4.

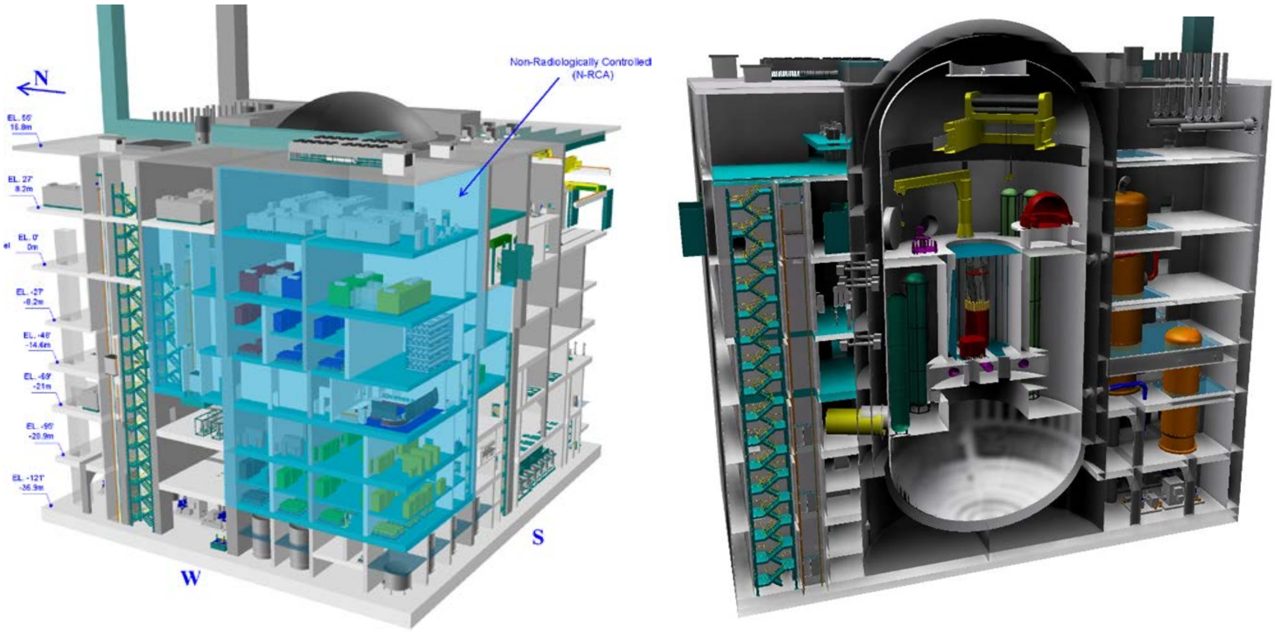


Figure 1.3. Nuclear island – external view and cut-through

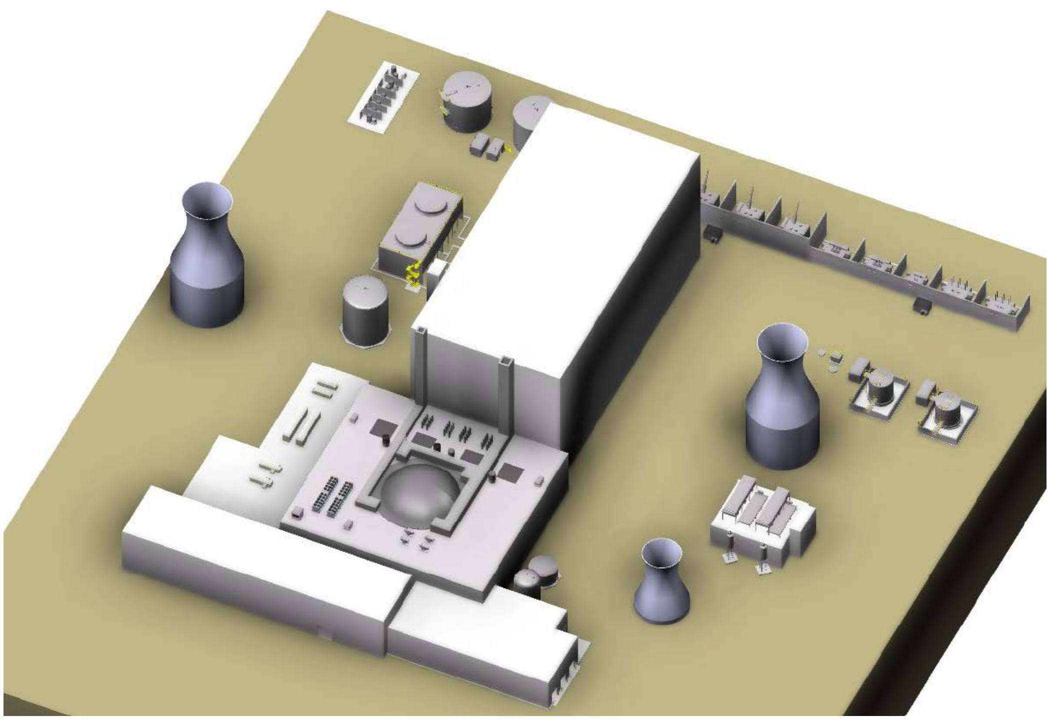


Figure 1.4. General Site Layout – 3D view

Notable safety features of the concept include:

- Fuel/clad system with enhanced accident tolerance
- Integral primary circuit (LB-LOCA and rod ejection eliminated)
- All safety systems are passive
- Passive core and spent fuel pool decay heat removal systems mitigates extended black-out
- Compact footprint and seismic isolators

These features enhance the safety beyond that of Gen-III+ systems. Specifically, the fuel/clad system is ultimately intended to be composed of high-conductivity and improved (higher) heavy-metal density silicide fuel (U_3Si_2) and low-oxidation rate SiC clad. However, recognizing that fuel qualification is a long end expensive process, the reactor is designed with an option (for lead units) to start operation on current licensed fuel (U_2O/Zirc), transition mid-term to U_3Si_2/FeCrAl, and long-term to U_3Si_2/SiC, as shown in Figure 1.5. Even with the current fuel, the overall system safety is enhanced due to other systems. Advanced steel (FeCrAl) cladding reduces the oxidation rate, as compared to Zirc-based materials, however it also introduces a non-trivial reactivity penalty. SiC cladding would further reduce the oxidation rate, while neutronically providing a similar performance as the Zirc-based cladding, but it is possible that its qualification may take longer than FeCrAl qualifications, since steel-based cladding (albeit with different steel type) was historically used in LWRs. This approach provides a flexible and viable path to deployment of improved fuel/clad system, since the initial reactor licensing and operation is not necessarily conditioned by the licensing of novel fuel.

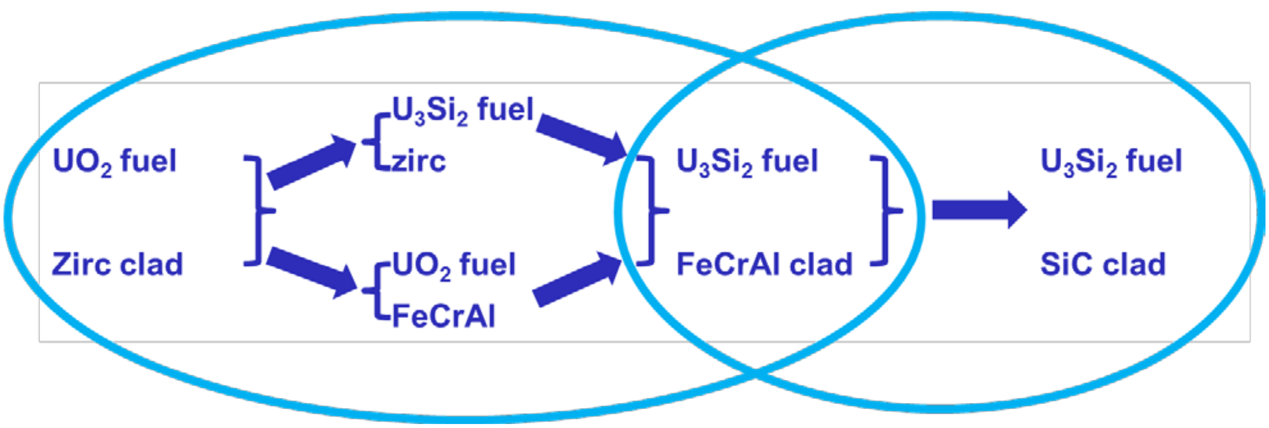


Figure 1.5. Flexible path to fuel with enhanced accident tolerance

Global aspects of the technical work were coordinated and harmonized at the whole-team team meetings (nine meetings held), while the detailed technical work occurred in technical working groups that were organized in all major technical areas.

The final outcome of the project is a fairly well developed concept of an advanced, integral PWR at 1,000 MWe power level, with enhanced safety and estimated competitive economics. A non-exhaustive list of main accomplishments follows:

- Harmonized the overall concept
- Applied holistic view to integrate design features, safety and economics
- Compiled a materials (fuel/clad) properties database
- Developed a framework for human-centered design approach and demonstrated the concept on the vessel design
- Selected an advanced fuel/clad system (U_3Si_2 fuel with FeCrAl or SiC cladding) with enhanced accident tolerance
- Developed a silicide swelling model, important for assessing viability of silicide fuel
- Performed selected – limited but relevant – experiments related to fuel and cladding
- Envisioned a viable path to novel fuel deployment
- Established the baseline fuel assembly design (19x19) and core layout (121 FA)
- Established the shut-down and control rod banks requirements
- Developed an I^{2}S-LWR core physics benchmark for cross-validation of core physics computational tools across the project
- Developed several options for the first core and equilibrium cycle and verified their acceptable performance
- Developed 2-batch and 3-batch refueling strategy with 18-month and 12-month refueling intervals
- Developed an advanced pseudo-equilibrium first core
- Performed fuel cycle cost analysis
- Evaluated Pu disposition capability of I^{2}S-LWR [performed and funded by, and of special interest to the UK team partner organization, aligned with the UK research priorities]
- Developed a detailed layout of the integral reactor vessel, with all primary components and internals
- Developed, to the appropriate level of detail, information on main pumps, integrated pressurizer, internal CRDMs, core barrel and radial reflector, automatic depressurization system)
- Performed comparative flow induced vibrations (FIV) analysis
- Evaluated thermal performance of the high power density core
- Performed preliminary vessel stress analysis
- Selected the micro-channel type heat exchangers (MCHX) for the primary heat exchangers and performed and optimized module design,
- Built MCHX experimental testing facility and performed relevant experiments
- Introduced and evaluated novel steam generation system (SGS) concept, based on the in-vessel single phase primary heat exchangers (PHX) and out-of-vessel flashing drums (FD)
- Developed and optimized Power Conversion System (PCS) based on the SGS concept to achieve target efficiency
- Established a detailed functional scheme of PCS
- Assessed potential benefits of an alternative PCS based on the Kalina cycle
- Established I&C strategy and main operational control algorithms for the core and flashing drums; performed system modeling and simulations; examined stability and self-diagnostics
- Examined I&C aspects of ex-core/in-vessel nuclear instrumentation, including novel routing
- Established safety philosophy

- Developed the concept and design of a passive decay heat removal system (P-DHRS), with ambient air as the ultimate heat sink; optimized its design
- Developed functional requirements and sized other safety systems (passive reactor cavity cooling system, passive containment cooling system, pressure suppression system, accumulators or makeup tanks)
- Developed a functional scheme of the containment with safety systems
- Developed a physical containment layout, considering operational, refueling and maintenance requirements
- Identified and classified relevant transient and accident scenarios
- Analyzed or assessed relevant transient and accident scenarios (LOFC, MFLB, MCHX blockage, SBO, SB-LOCA, SPADS, RIA. MSLB, inadvertent DHRS actuation)
- Performed Level 1 PRA and used results to guide design modifications to improve CDF
- Estimated CDF of the optimized system; confirmed it met the target requirements
- Performed preliminary PRA uncertainty qualification
- Established used nuclear fuel (UNF) management approach
- Evaluated I²S-LWR fuel decay heat characteristics
- Developed the spent fuel pool concept with a P-DHRS, with ambient air as the ultimate heat sink; optimized its design
- Developed a fast simulation tool for enhanced safety and security monitoring of SFP
- Evaluated fast neutron fluence on reactor vessel to assess its lifetime
- Evaluated and optimized type and placement of the ex-core/in-vessel nuclear instrumentation detectors
- Evaluated activation of MCHX to inform maintenance activities
- Evaluated dose distribution inside and outside the containment vessel to inform radiation protection
- Evaluated gamma heating of the radial reflector and assessed cooling needs (cooling channels)
- Developed nuclear island layout
- Established the nuclear island seismic isolation concept
- Developed the NPP site layout
- Established the differential economics analysis framework
- Performed differential economics analysis and assessed economic competitiveness
- Identified potential path(s) forward
- Identified important outcomes of this project with potential broader positive impacts beyond this project
- Educated and trained over 150 students and young researchers
- Engaged senior-level students in 14 senior design projects related to I²S-LWR
- Documented results in reports (14 quarterly reports; over 3300 pages total) and over 60 peer-reviewed papers published in journals and conference proceedings

Overall, I²S-LWR has been a very successful project, with some of its findings potentially having a significant impact beyond the project itself, on future trends in LWR technologies.

2. Roadmap through the Final Report and Topical Reports

The IRP Project "Integral Inherently Safe Light Water Reactor" is a large, multidisciplinary project—addressing many technical areas—that generated a wealth of new results. Therefore, it has been quite challenging to capture all information in an effective format.

The solution is found in the following approach. This "Final Report" provides a high-level summary (expanded with further details when deemed necessary or beneficial), while relegating more detailed technical descriptions to topical reports. The final report is the final formal deliverable of the project. The topical reports generally aim to capture the work covered in quarterly reports, and to present it in a better organized form for future reference. The 14 quarterly reports combined amount to over 3,300 pages, and to about 4,000 pages with graduate thesis, . Results of different tasks are presented incrementally, as the work progressed, as well as iteratively, i.e., with changes occurring over time and sometimes making the early results obsolete. Thus, while all results are documented in quarterly reports, extracting results on a specific topic requires non-trivial efforts. The topical reports aim to bridge that gap and collect the relevant final results on a topic-by-topic basis.

Topical reports were not listed/required as final deliverables, but the team felt that this is the best way to document significant achievements accomplished by this project. However, preparation of the topical reports took significant additional time and efforts. Eventually, 15 topical reports have been prepared, covering specific topics, with about 1,500 pages total. Compared to the quarterly reports and theses, the volume has been reduced almost three times, while preserving essentially all technical information. Moreover, the organization by topic will make the future access to information immensely more practical and effective.

The final report and topical reports are organized as follows:

- Chapter 0: Executive summary. It includes the basic logic and ideas of the I^{2}S-LWR concept, top-level requirements, approaches and solutions, and a list of main achievements.
- Chapter1: Roadmap through the final report and topical reports (this chapter). Intended to help the reader to locate the information he needs.

Followed by Part I, presenting the fundamentals of the I^{2}S-LWR Concept, with details on fuel/clad system selection and high power density core performance. These two specific topics are covered here in more detail than other topics since they address the question of technical viability of the concept,

- Chapter 2: Programmatic summary of the project.
- Chapter 3: I^{2}S-LWR Concept Overview. Captures in one place, with some detail, main characteristics, layout, solutions, parameters, and performance parameters of I^{2}S-LWR.
- Chapter 4: Fuel with Enhanced Accident Tolerance – Basic Considerations. Presents the main ideas and approaches to viable introduction of ATF into the LWR fleet through I^{2}S-LWR. Summarizes design choices considered for fuel and cladding in I^{2}S-LWR, including the challenges and a potential show-stopper (swelling) for silicide fuel. Proposes a flexible path to commercial

deployment. While this is covered in more details in the topical reports on Materials and on Core Design and Performance, it was deemed important to present the main facts in this overview.

- Chapter 5: High Power Density (HPD) Core Thermal Performance Assessment. Together with ATF, HPD core is one of the pillars of the I²S-LWR concepts. Therefore, it was deemed important to summarize here the assessment of its viability, even though most of this analysis is repeated in the corresponding topical report on Core Design and performance.

Subsequent Chapters in Part II provide in one place executive summaries of individual topical reports, thus facilitating locating typically similar to the executive summary of that topical report, for major topics, including:

- Materials
- Core Design and Performance
- Thorium-based Plutonium Incineration
- Reactor Vessel Layout and Internals
- Power Conversion System
- Safety and Transient Analyses
- Probabilistic Risk Assessment (PRA)
- Instrumentation and Control
- Shielding Analyses
- Containment Layout
- Plant Layout
- Spent Fuel Pool Analysis: Passive Decay Heat Removal System
- Spent Fuel Pool Analysis: Subcriticality Monitoring and Safeguards
- Economics

Chapters in Part III include:

- Path forward
- Conclusions and recommendations

Part IV contains Appendices.

Part I – I^2S-LWR Concept

3. I²S-LWR Project

3.1 Project Overarching Objective and Performance Period

The Integral Inherently Safe Light Water Reactors (I²S-LWR) Concept proposal was submitted to DEO NEUP in 2012, awarded decision made in September that year. The project research was performed during the period 2/2013-12/2016. In response to DOE solicitation, the proposed goal of the project was to develop a concept of a 1 GWe PWR (for mainstream US applications) with inherent safety features beyond those of Gen-III+ systems, and accounting for lessons learned from the Fukushima accident. At the same time, economic viability of the new concept should aim to be similar to or better than that of Gen-III+ systems. Details are elaborated upon in the remaining part of this report, but essentially, the high level approach was to extend attractive safety features of small modular reactors (SMRs), in particular integral configuration, to a higher power level design, and combine it with enhanced accident tolerance. This however requires novel solutions and novel technologies.

3.2 Project Team

An expert, multi-disciplinary and multi-organization team was formed, initially composed of 11 organizations. There organizations joined during the course of the project. This expanded team of 14 organizations, lead by Georgia Tech, included seven other US universities (Brigham Young University, Florida Institute of Technology, University of Idaho, University of Michigan, Morehouse College, University of Tennessee, and Virginia Tech), nuclear industry and utility (Westinghouse Electric Co. LLC and Southern Nuclear), national laboratory (Idaho National Laboratory), and three international academia partners (University of Cambridge, UK; Politecnico di Milano, Italy; and, University of Zagreb, Croatia). This diverse expert team ensured successful completion of the project, while the participation of industry provided invaluable practical expertise and sanity-check throughout the course of the project.

Team member organizations with their abbreviations and senior personnel with their initials (used for identification in this document) are given in Table 3-1.

Table 3-1: Team member organizations with their abbreviations and senior personnel with initials their used for identification in this document

Team member organization	**Abbrev.**	**Personnel and initials**
ACADEMIA		
Georgia Institute of Technology	GT	Bojan Petrovic (BP) PI Farzad Rahnema (FR) Co-PI Chaitanya Deo (CD) Srinivas Garimella (SG) Preet Singh (PS) Glenn Sjoden (GS) Dingkang Zhang (DZ)
University of Michigan	UM	Annalisa Manera (AM), Co-PI Thomas Downar (TD) John Lee (JL)

University of Idaho	UI	Indrajit Charit (IC)
University of Tennessee	UT	Belle Upadhyaya (BU), Co-PI J. Wesley Hines (WH)
Virginia Tech	VT	Alireza Haghighat (AH) Co-PI
Brigham Young University	BYU	Matthew Memmott (MM) Co-PI
Florida Institute of Technology	FIT	Guy Boy (GB) Co-PI
Morehouse College	MC	Lycurgus Muldrow (LM) Co-PI
INDUSTRY		
Westinghouse Electric Company	WEC	Paolo Ferroni (PF) Co-PI Fausto Franceschini (FF), Matthew Memmott (MM), David Salazar (DS), William Mack (WM), Jason Young (JY), Alex Harkness (AH), Robert Ammerman (RA), Matthew Smith (MS)
Southern Nuclear Company	SNC	Ronald Cocherell (RC) Co-PI (2013-14) Nick Irvin (NI) Co-PI (2015-16) Nick Smith
NATIONAL LABORATORIES		
Idaho National Laboratory	INL	Abderrafi Ougouag (AO) Co-PI George Griffith (GG)
INTERNATIONAL COLLABORATION		
University of Cambridge, UK	UCA	Geoffrey Parks (GP) Co-PI
Politecnico di Milano, Italy	POLIMI	Marco Ricotti (MR) Co-PI
University of Zagreb, Croatia	FER	Nikola Čavlina (NC), Davor Grgić (DG), Dubravko Pevec(DP)(Co-PIs)
CONSULTANTS		
	HG	Hans Garkisch (HG)

In addition to over 30 Co-PIs and senior team members listed in Table 3-1, the project engaged 11 young faculty, researchers, scientists and post-docs, as well as over 40 graduate (MS and PhD) students, and over 100 undergraduate students, most of them through senior design projects. Thus, more than 150 young faculty/researchers and students were trained and had opportunity to work on a cutting-edge research, under realistic real-life R&D conditions. This education and training by itself provides an excellent "return on investment" to DOE.

A list of all contributors and participants is given in Appendix B. This list aims to give credit to all contributors to the project during the whole project duration. It does not imply that all contributors were fully funded or devoted full time to the project. Some contributors (some graduate students) were fully funded, some contributors were partially funded, while others were funded through a fellowship or a synergistic activity and thus contributed at no cost to the project. Similarly, the list does not aim to quantify the scope of individual contributions. It ranged from a short-term expert help, or a one-semester-undergraduate research project, to multi-year full time graduate research.

Table 3-2: Responsibilities of team member organizations

Lead or Coordinator	Area
GA Tech	Project management and integration; overall system design Fuel and cladding materials, fuel assembly design (with WEC) Core design and fuel cycle (with WEC) Primary Heat Exchangers and Test Facility Shielding and activation analysis Proliferation Resistance and Safeguard
Brigham Young U.	Components design, contribute to safety analyses (with UMich)
FL Tech	Tools for effective project integration (human centered design)
INL	Silicide fuel swelling model
Morehouse	Educational aspect / outreach
POLIMI	Economic analyses (with GT). Student exchange.
Southern	Utility perspective
U. Cambridge	Thorium cycle option
U. Idaho	Cladding materials testing
U. Michigan	Design of components, thermal and safety systems (with WEC) Coupled neutronics/thermal-hydraulics analyses System and safety analyses (deterministic and PRA/DPRA)
U. Tennessee	I&C
U. Zagreb	Contribute to core analyses and economics
VA Tech	Spent fuel storage and monitoring
WEC	Industry perspective and expertise: Fuel and cladding materials, fuel assembly design (with GT) HPD core performance, core design and fuel cycle (with GT) Design of components, thermal and safety systems (with UMich) O&M, BOP and Layout, Seismic

3.3 Team Members Responsibilities

Clear responsibilities were established among the team members to lead and coordinate specific project areas, as shown in Table 3-2. However, the team performed in a tightly-coordinated manner, with contributions of multiple team members to most areas, i.e., the team members contributed to other areas in addition to those listed in Table 3-2.

3.4 External Advisory Board (EAB)

The External Advisory Board (EAB) was established at the very beginning of the project. The EAB consisted of experienced senior executives:

- Regis Matzie (Panel Chair), retired Chief Technology Officer (CTO) of Westinghouse Electric Co.;
- John McGaha, retired Senior Executive of Entergy Nuclear;
- Ted Marston, retired CTO of EPRI and Principal of Marston Consulting;
- Rosa Yung, Vice President Innovation, Electric Power Research Institute; (2013-2015)
- Albert Machiels, Senior Technical Executive, Electric Power Research Institute; (2015-2016)
- Lynn Weaver, retired President of Florida Institute of Technology; and,
- Chuck Kling, retired Consulting Engineering of Westinghouse Electric Co.

The EAB met twice a year and provided feedback, questions, and invaluable industry perspective to the team in a form of letter report.

3.5 Reporting

Results have been documented in 14 quarterly reports as summarized in Table 3-3, totaling over 3,300 pages. Therefore, it is impossible repeating all the details in this final report. Instead, it aims to capture main findings and results.

All reports have been uploaded to PICS. The reports have been carefully reviewed by the DOE Technical Monitor, Don Williams (ORNL) and Federal Program Manager, Damian Peko. Their questions and comments contributed to maintaining the quality and resolving issues that appeared during the course of the project.

3.6 Project Status Presentations to DOE

Regular annual briefings on the project status were given to the DOE leadership:

- Presentation to Peter Lyons and staff, Georgia Tech, Atlanta, GA, June 17, 2013
- Presentation to DOE at the 2nd Team Meeting, Arlington, VA, December 12, 2013
- Presentation at DOE, Germantown, MD, May 7, 2015
- Presentation at DOE HQ, Washington, DC, May 12, 2016

Additionally, the projects was presented at several NRT and AFC integration meetings, where a strong synergy with the DOE ATF efforts was observed.

Table 3-3: Team member organizations with their abbreviations and senior personnel with initials their used for identification in this document

FY	QTR	Vol's/Rev	Pages
2013	Q2	Vol.1 r1	51
	Q3	Vol.1 r0	94
		Vol.2 r0	327
	Q4	Vol.1 r1	142
		Vol.2 r1	331
		Vol.3 r1	118
2014	Q1	Vol.1 r0	115
		Vol.2 r0	185
	Q2	Vol.1 r1	129
	Q3	Vol.1 r0	199
	Q4	Vol.1 r1	236
2015	Q1	Vol.1 r0	214
	Q2	Vol.1 r0	210
	Q3	Vol.1 r1	175
	Q4	Vol.1 r1	183
2016	Q1	Vol.1 r2	291
	Q2	Vol.1 r1	125
	Q3	Vol.1 r2	164
	Q4	Vol.1 r0	13
		TOTAL	3302

3.7 Dissemination of Project Results

Results have been disseminated through journal articles and peer reviewed conference papers (about 30 each), extended summaries (over 10), as well as presentations at conferences and technical seminars. Additionally, presentation of the project to a broader audience has been achieved through: (a) organizing a special panel session at the 2014 ANS Winter meeting (presenting the initial results); and (b) organizing and editing two volumes of a special issue of Annals of Nuclear Energy devoted exclusively to I²S-LWR to capture main results and findings. (Volume 1 was published in 2/2017; Volume 2 expected by end 2017.) List of publications is provided in Appendix C.

3.8 Senior Design Projects Related to I²S-LWR Project

The project devoted significant efforts to engage undergraduate students, primarily through senor design projects, but also through individual research projects. Over 100 undergraduate students participated in this activity which proved to be a true win-win situation. The students were trained and obtained a real-life experience of participating in a cutting edge multidisciplinary project with all its challenges and complexities. Ultimately, they were rewarded by knowing that this was not just a formal class work, but in fact they contributed in a meaningful way. Reversely, the project benefited from the student design projects.

It is in particular worth mentioning the initial boost given to the project by 7 student design projects started in January 2013, even before the formal start of the project (February 2013 for Georgia Tech, and additional few months for other team members due to subcontracts signing). Thus, while the initial contractual interactions were being resolved and formal work was just ramping up, the project made significant progress due to these senior design projects. With the help and advisement of faculty and Westinghouse experts, these projects in fact developed the detailed 3D layout of the reactor vessel with all internal components (updated later, but the bulk remained this initial work), and established the basis for future work on internal microchannel heat exchangers as a basis for the steam generation system, among other contributions. The projects served to introduce the students to novel technologies, e.g., additive manufacturing. A 3-ft tall reactor vessel model printed initially proved to be a great tool to understanding the intricacies of the integral design. At the endo of the project, a 3D models of the containment vessel, nuclear island building, and the whole site were printed as well.

While most of the senior design projects occurred at Georgia Tech, students were also engaged in this type of activity at BYU, U. of Tennessee and U. of Idaho. Figure 3.1 shows examples of final posters prepared for several senior design projects. Appendix C provides a list of all the senior (undergraduate) and advanced (graduate) design projects.

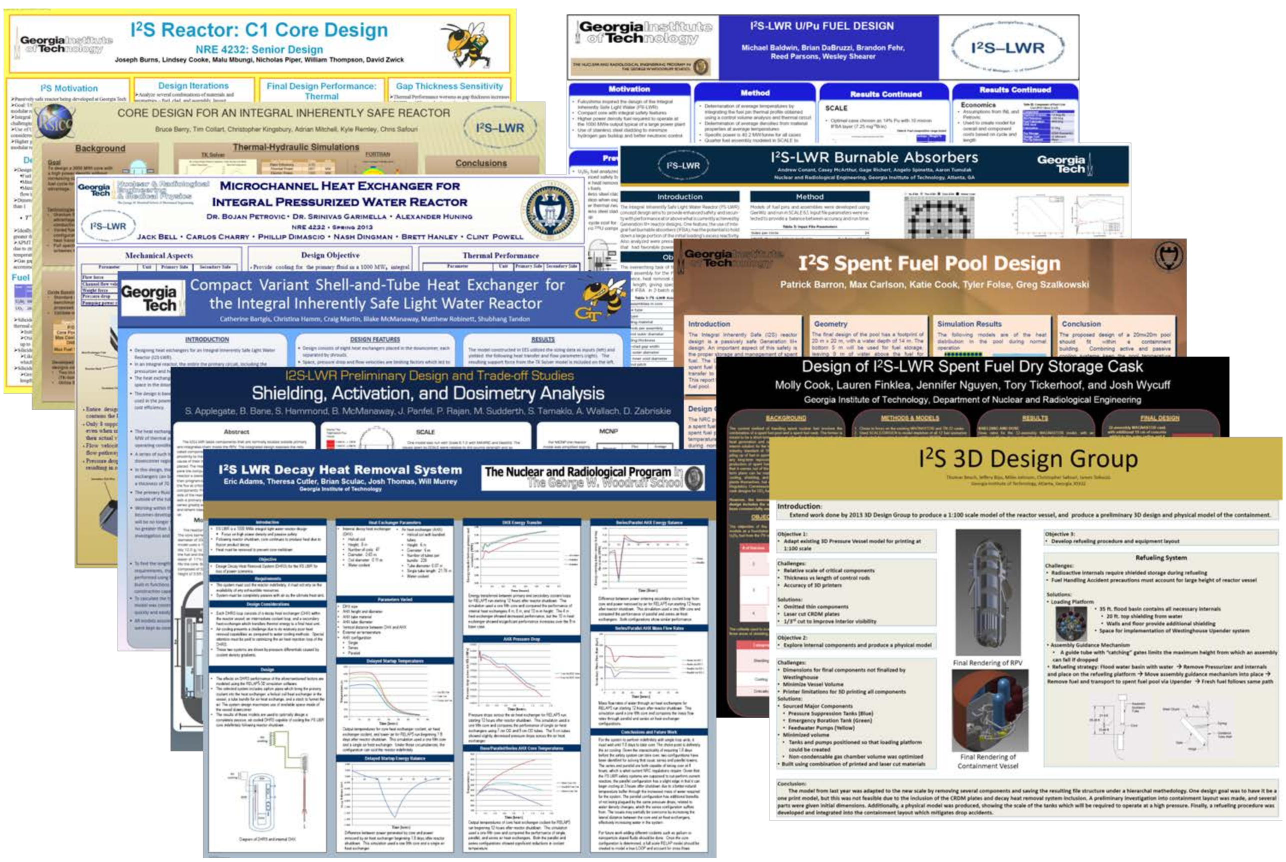


Figure 3.1. Sample posters prepared for senior design projects

4. I²S-LWR Concept Overview

4.1 Overarching objectives and main features

The overarching objective of the I²S-LWR concept is to extend attractive safety features of SMRs, in particular integral primary circuit configuration, to a higher power level, around 1 GWe, and thus offer a LWR with enhanced safety to those markets preferring larger power units, potentially as the next step beyond the Gen-III+ reactors (Figure 4-1). We should explicitly state that this does not imply a preference for large (or small) units; instead, it aims to offer an attractive new option to the mainstream power market.

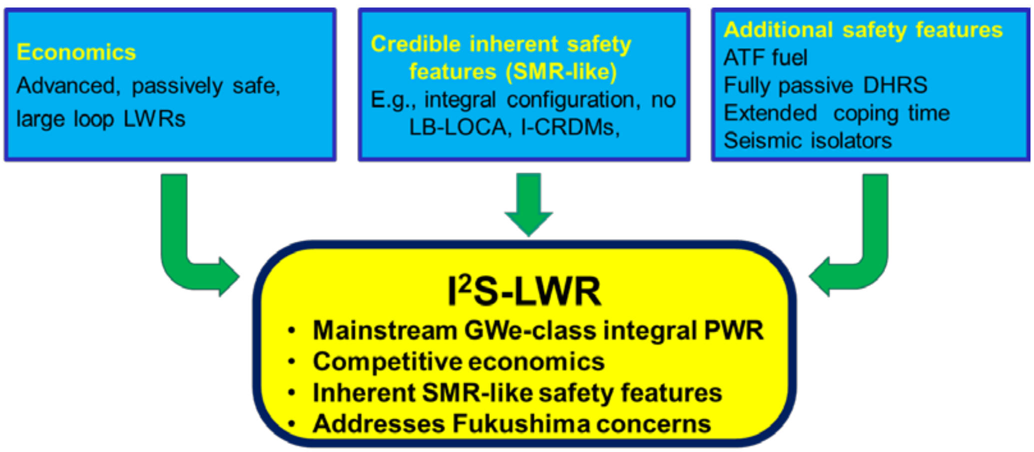


Figure 4-1. I²S-LWR concept

Additional specific techno-economical goals are to address post-Fukushima concerns and apply lessons learned, while devising a concept with potential for economic competitiveness with respect to Gen-III+ designs. Main features contributing to the safety are:

- Safety-by-Design, aiming to eliminate the initiators of as many accidents as practical;
- Integral configuration, fully eliminating some of the most challenging accidents (e.g., LB-LOCA and rod ejection);
- Fuel with enhanced accident tolerance (thus terminating or delaying progression of some accidents)
- Passive safety systems
- Fully passive decay heat removal system (P-DHRS), with ambient air as the ultimate heat sink, thus providing indefinite cooling capability (no need to replenish a pool) in many accident scenarios, with minimum or no operator action required;
- Similar P-DHRS for spent fuel pool (addressing post-Fukushima concerns);eliminating ;
- Compact nuclear island, on seismic isolators, reducing impact of seismic events

4.2 Approach to the I²S-LWR concept development

An integral LWR configuration at that 1 GWe power level is not achievable with current technologies; this is one of the reasons that integral SMR concepts have been limited to no more than 300-350 MWe.

The I²S-LWR concept therefore needs to develop novel technologies, approaches and solutions to enable the several times higher power output. Referring to the I²S-LWR schematic in Figure 4-2, new technologies needed, as compared to current loop PWRs, essentially fall into two categories.

1. I²S-LWR specific, novel technologies
    a. High power density fuel/clad system with enhanced accident tolerance
    b. High power density (micro-channel type) primary HX (MCHX)
    c. Steam Generation System (MCHX + Flashing Drum)
    d. Passive safety systems (conceptually similar to some of the existing systems, but with specific features, e.g., air as the ultimate heat sink in P-DHRS)

2. Technologies being developed for integral LWR SMRs:
    a. Integral layout
    b. Integral primary components (e.g., I-CRDMs)
    c. Instrumentation for integral PWRs

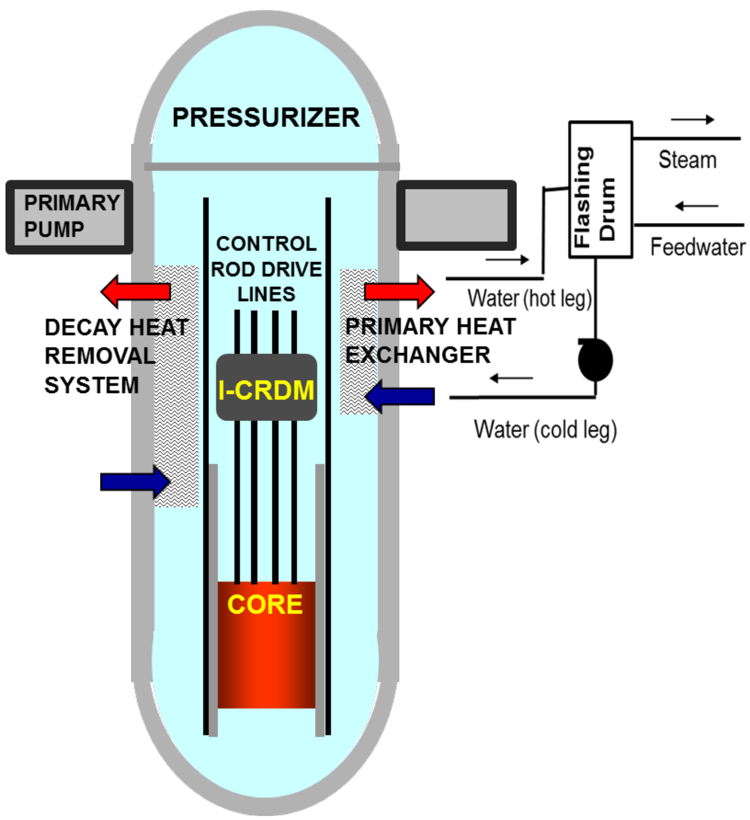


Figure 4-2. I²S-primary circuit and steam generation system (SGS) schematic

Clearly, the first category was the focus of the research, while the second category focused on the specific adaptations, if needed. For example, the integral vessel layout incorporates many general features found in integral SMRs (and in particular IRIS and WEC SMR), but it is significantly adapted to meet the I²S-LWR requirements. On the other hand, Westinghouse developed and partly tested internal control rod drive mechanism (I-CRDM) for WEC SMR, and positively assessed its feasibility. The Project therefore assumed that if/when needed I-CRDMs can be developed, and that further development at this stage is not needed, i.e., would only divert limited Project resources.

4.3 Top level requirements

The project was guided by the top level requirements (Table 4-1) that were established in the proposal, and updated and somewhat expanded during the early project phase. The requirements were formulated in terms of hard 'must satisfy' requirements, with additional soft 'it would be valuable to satisfy' stretch targets. Essentially all hard requirements have been met; in addition, some of the stretch targets have been met as well. Last column in Table 4-1 summarizes how the final design met the requirements.

Over the first project year, trade-off studies were performed and down-selections made, followed by establishing the preliminary concept, fairly detailed for the stage of the project, but still lacking design information in certain areas. During the second project year, most of the missing areas and details were covered resulting in a modified concept. Critical review over the third project year led to further improvements and harmonization, resulting in the final concept. Potential path forward was also addressed at the end of the project.

Table 4-1: Top level requirements.

	Requirement	**Stretch Target**	**Comment**	**Final**
APPLICATION-DRIVEN REQUIREMENTS				
Power	~1,000 MWe		For markets preferring large plants; economy of scale	985 MWe
Electricity production efficiency	>32%	35%	Economic competitiveness; reduced reject heat	>34%
Design lifetime	60 years	100 years	Competitiveness; economics, sustainability	100 years
Reactor pressure vessel	Same size as or smaller than current large PWRs (e.g., EPR, ABWR) [~5m]		Manufacturability	490 CM I.D.
FUEL-RELATED REQUIREMENTS				
Fuel/cladding system	Enhanced accident tolerance		Post-Fukushima considerations	Clad: reduced oxidation rate Fuel: high conductivity
Fuel enrichment	Viable reloading with <5% enriched fuel	Potentially improved fuel cycle with 5-8% enriched fuel	Possibility to use existing infrastructure for <5% enrichment Stretch target abandoned due to reduced industry interest for >5% enriched fuel	Licensed <5% enriched fuel

Refueling	Multi-batch, refueling interval 12 months or longer	Options for 12-18-24 months refueling	24-mo cycle when required by utilities 24-mo cycle possible but may require >5% enr for competitive FCC (reduced industry interest)	12-month and 18-month refueling scheme developed
SAFETY AND SECURITY				
Security, safeguards and proliferation resistance	As in current LWRs or better		Compact partly under grade nuclear island	Low profile; improved physical protection
Safety indicators	CDF <3x10^{-7} LERF <3x10^{-8}	CDF <1x10^{-7} LERF <1x10^{-8}	Improve safety indicators relative to current Gen-III+ passive plants	Preliminary PRA meets stretch target
Safety philosophy/systems	Inherent safety features Full passive safety High level of passivity		Eliminate accident initiators. Eliminate need for offsite power in accidents	Full passive safety. LB-LOCA and rod ejection eliminated.
Grace period	At least 1-week	Indefinite for high percentage of considered scenarios, with no/minimum operator action	Resistance to LOOP and Fukushima-type scenarios	At least 1 week. Indefinite in many scenarios.
Decay heat removal	Passive system with air as the ultimate heat sink		Resistance to LOOP and Fukushima-type scenarios	Passive 3-of-4 DHRS-to-air designed
Seismic design	Single compact building design	Seismic isolators	Allows siting at many locations	Compact footprint, Seismic isolators
Other natural events	Robust design		Address unforeseen events	Robust design
Monitoring	Enhanced, in normal and off-normal conditions		Improve normal operation; Address unforeseen events	Robust self-diag I&C algorithms
Spent fuel pool safety	Monitoring Passive cooling		Address Fukushima issues with SFP	Passive air-cooled DHRS
Used nuclear fuel management	On-site, for the life of the plant		Remove reliance on repository availability at certain date	10 year SFP pool and dry cask storage.
DEPLOYMENT REQUIREMENTS				
Economics	Competitive with current LWRs		Differential economics	Diff. econ. indicates competitive
Deployment	Near-term: feasible with <5% enriched oxide fuel	Long-term option: up to 8% enriched if industry interest	Oxide fuel provides path to accelerated deployment	Option to start with current fuel; mid-term FeCrAl and

				U3Si2; long-term SiC clad
Operational flexibility	2-batch and 3-batch, ≥12-month cycle	5% and 8% 12-18-24 months cycle	Diverse market needs Currently reduced interest for >5% enrichment fuel; therefore not developed	Focus on <5% fuel for now.
Operational flexibility		Load follow with MSHIM	Reduced effluents (environmental)	MSHIM assessed feasible but not developed in detail
D&D	Returned to green site simplified		Sustainability and public acceptance	Reduced activation outside RPV simplifies D&D and reduces dose

The main plant parameters are given in Table 4.2. I²S-LWR reactor vessel (RV or RPV) is shown in Figure 4.3 (schematic with the steam generation system, radial cut, external view and cut-through view to show internals).

Table 4.2: I²S-LWR main parameters.

PARAMETER	VALUE
Thermal Power	2850 MWth
Electric output (net)	~1000 MWe (986 MWe for the considered environmental temp.)
Net thermal efficiency	34.6%
Vessel inner clad I.D.	490 cm
Vessel height	22.87 m
Primary circuit	Integral configuration
Primary pressure	2250 psi
Core	121 fuel assemblies
Fuel assemblies	19x19 lattice (336 fuel rods, 24 control rod guides, 1 central instrumentation guide tube)
Active core height	12 ft
Enrichment	<5%
Refueling	18- and 12-month; 2- and 3-batch
Reactor coolant pumps	8
Primary heat exchangers (PHE)	Microchannel type (MCHX), 8 modules, paired in 4 subsystems
Steam generating system	PHE and flashing drums
Decay heat removal system	Fully passive, 4 trains, ambient air ultimate heat sink
Containment O.D.	23 m
Containment height	52 m
Plant design life	>100 years (based on reactor vessel fluence)

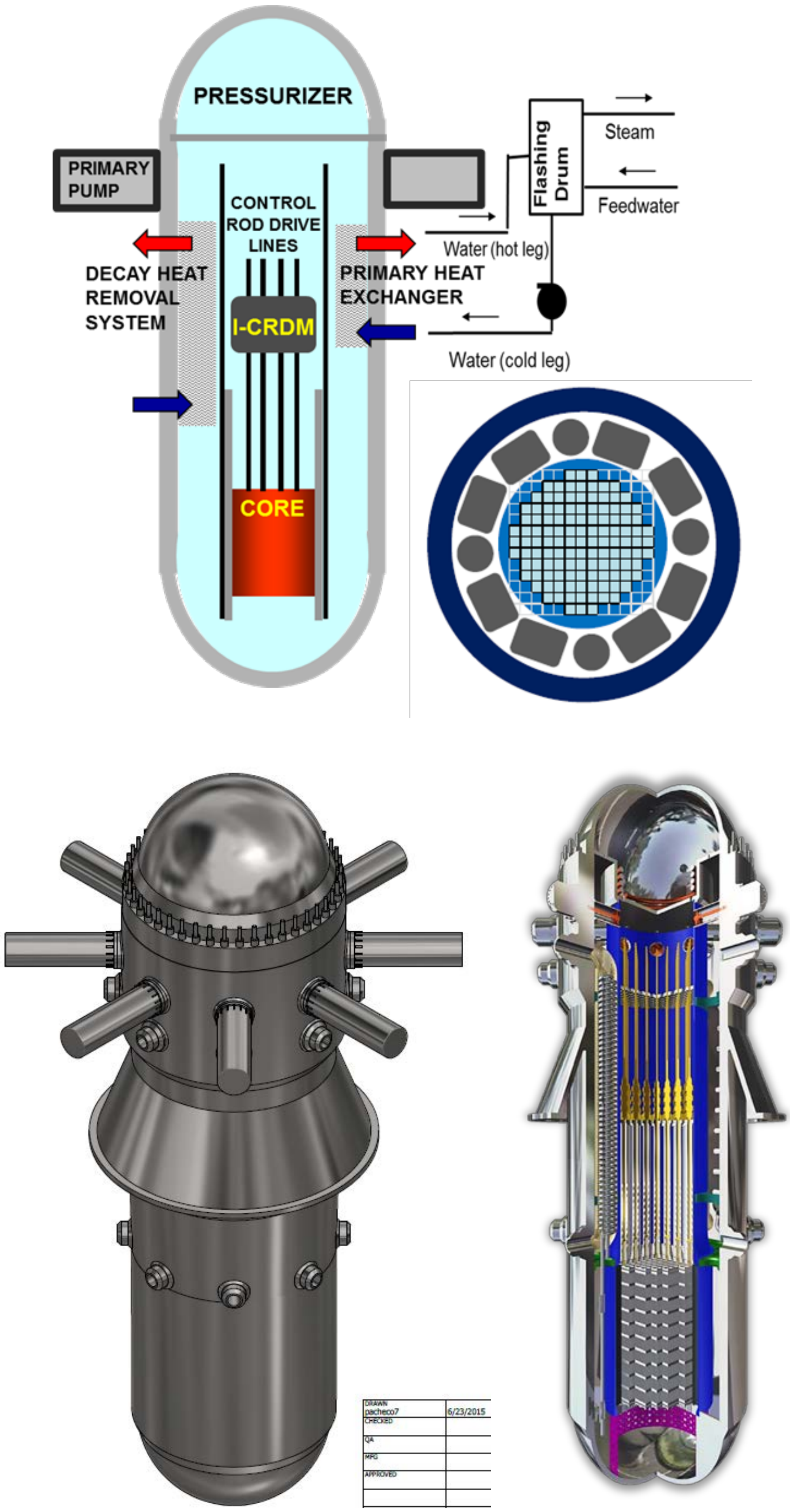


Figure 4.3. Reactor vessel. Top row: schematic, vertical and radial cut.
Bottom row: 3D model external view and cut-through

I²S-LWR integral configuration is characterized by:

- All primary circuit components are located within the RPV
- Primary coolant circulates and remains within the RPV
- Only the secondary loop (single phase liquid) enters/exits reactor vessel

All primary circuit components are located within the RPV, including:

- Core
- Radial reflector
- Control rods and control rod drive mechanism
- Internals (barrel – upper and lower, core support, control rod boxes)
- Pressurizer integrated in the upper vessel head
- Primary heat exchangers (MCHX, used to remove heat in normal operation)
- Decay heat removal heat exchangers (redundant and diverse way of removing decay heat)
- Seal-less reactor coolant pumps attached to the vessel
- Automatic depressurization system at the vessel head

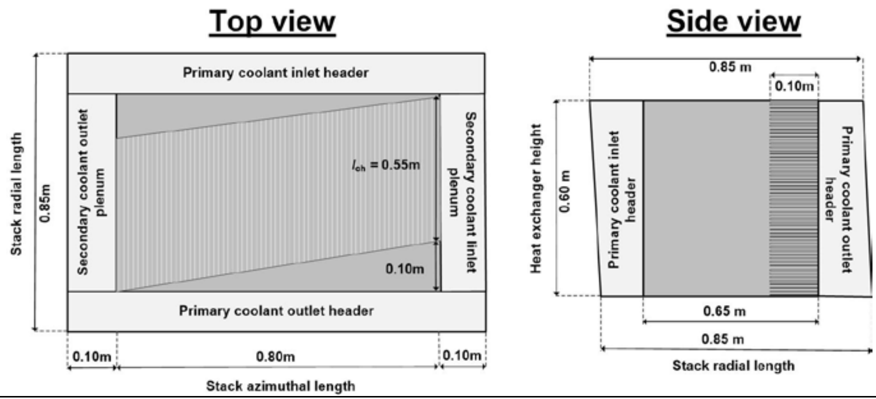
Note that the steam generating system (SGS) contains primary heat exchangers inside the vessel, and flashing drums outside the vessel. It has 4 trains with 2 paired modules each. Similarly, the fully passive emergency decay heat removal system (P-DHRS) has four heat exchangers (four trains) inside the vessel, and towers (heat sink) outside the containment.

More details on individual systems, component and analyses are provided in subsequent chapters. However, a summary of various plant parameters is given in Table 4.3. Most parameters are listed in SI units. Some parameters are also listed in non-metric units to facilitate comparison.

Table 4.3: Plant parameters.

General Parameters		
NSSS power	MWt	2,850
Target net electric power	MWe	1,000
Reference design net electric power [MWe]	MWe	986
Net thermal efficiency	%	34.6
Reactor coolant pressure, operating	MPa (psi)	15.513 (2,250)
Average core inlet temperature	°C (°F)	298 (568.4)
Average core outlet temperature	°C (°F)	330 (626)
Reference plant design life	Years	60
Plant design life (based on reactor vessel fluence)	Years	>100
Reactor Vessel and Internals		
Configuration		Integral
Reactor vessel clad I.D.	m	4.900
Reactor vessel clad thickness	m	0.008
Reactor vessel base material.		SA-508
Reactor vessel base metal I.D.	m	4.916

Reactor vessel base metal thickness (cylindrical portion)	m	0.250
Reactor vessel base metal thickness (hemispherical portion)	m	0.125
Reactor vessel O.D.	m	5.416
Reactor vessel height external	m	22.779
Reactor vessel height internal	m	22.529
Core equivalent diameter	m	2.867
Radial neutron reflector O.D. (NOTE: thicker reflector possible)	m	3.20
Lower core barrel I.D.	m	3.20
Lower core barrel O.D.	m	3.30
Upper core barrel I.D.	m	2.90
Upper core barrel O.D.	m	3.00
Downcomer annulus width (at PHE elevation)	m	0.95
Reactor vessel design temperature	°C (°F)	343.3 (650)
Reactor vessel fast (>1MeV) neutron fluence over lifetime	n/cm^2	$<1x10^{19}$ (100 yr)
Reactor Coolant Pumps (RCP)		
RCP type		Seal-less
Number of RCP		8
Pump speed	RPM	1,400
Impeller diameter	m (in)	0.734 (28.9)
Pump pressure head	psi	115.2
Efficiency (%)	%	87.0
Motor power	MW (HP)	2.54 (3,406)
Total power to coolant (per 8 pumps)	MW (HP)	20.32 (27,248)
Motor frame length	m (in)	2.794 (110)
Estimated weight (per pump)	kg	6,000
Pressurizer (PZR)		
Number of units		1
Type		Integrated in upper vessel head
Height	m	4.97
Total volume	m^3	54.62
Water volume	m^3	42.50
Primary Heat Exchangers (PHE)		
Type		Micro-channel (MCXH)
Mode of operation		Liquid-liquid
Number of subsystems		4
Number of modules per subsystem		2
Total number of modules		8
Number of units per module		11
Total number of units		88
Single unit shown below		



MCHX stack radial length	m	0.85
MCHX unit radial length	m	0.65
Primary coolant header radial length	m	0.20
MCHX stack azimuthal length	m	1.00
MCHX unit azimuthal length	m	0.80
Secondary coolant plenum azimuthal length	m	0.20
MCHX unit height	m	0.60
Active channel length, l_{ch}	m	0.55
Channels per sheet		445
Sheets per MCHX unit (total)		530
Primary side coolant flow rate (including bypass)	kg/s	15,498
Primary side coolant flow rate (excluding bypass)	kg/s	14,723
Secondary side coolant flow rate	kg/s	13,016
Secondary side coolant Tin (into MCHX)	°C (°F)	279.3
Secondary side coolant Tout (out of MCHX)	°C (°F)	318.2
Maximum allowed pressure drop across the MCHX	MPa	0.500
Secondary coolant Pin (into MCHX)	MPa	16.30
Secondary coolant Pout (out of MCHX)	MPa	15.84
Fouling allowance, deposit layer thickness	μm	10
Flash (Steam) Drums (FD)		
Type		Horizontal inflow; mesh separator
Number of subsystems		4
Number of drums		4
Flow recirculation	%	~90
Drum O.D.	m	5.72
Drum shell thickness	m	0.076
Drum height	m	20.172
Steam Generating System (SGS)		
Type		Liquid-liquid in-vessel MCHX PHE and out-of-vessel FD

Overall net efficiency	%	34.6
Containment Vessel (CV)		
Type		Cylindrical steel containment
Material		SA 738B
Containment O.D.	m	23.000
Containment height	m	52.000
Thickness	m	0.06 (2.375”)
Maximum design pressure	MPa (psi)	0.9 (130)
Safety systems within the containment		
Pressure Suppression system (PSS)		
Number of PSS tanks		4
Volume of each PSS tank	m^3	402
Accumulators (ACC)		
Number of ACC tanks		4
Volume of each ACC tank	m^3	271
Core Makeup Tanks (CMT)		
Number of CMT tanks		2
Volume of each CMT tank	m^3	151
Passive Reactor Cavity Cooling System (PRCCS)		
HX length	m	8
HX height	m	0.7
Passive Containment Cooling System (PCCS)		
HX total tube length (helical configuration)	m	4,000
Tube diameter	m^3	0.05
Passive Decay Heat Removal System (P-DHRS)		
Type		Fully passive
Ultimate heat sink		Ambient air
Heat sink capacity		Unlimited
Configuration		Internal liquid-liquid XH; external tower with liquid-air HX
Number of trains		4
Each train capability relative to the requirement	%	34
Redundancy		3-out-of-4
Tower height	m	30
Number of towers		2
Plant Layout		
Nuclear Island (NI) dimensions, footprint, including SGS NOTES: Not fully optimized. Reduction by ~10-20% in each dimension deemed realistic.	m^2	63x69

Nuclear Island height, top of the lowest floor to top of the roof	m	53.7
Nuclear Island total height, from the basemat bottom	m	62
Top of the lowest floor elevation (below grade)	m	-36.9
Top of the roof elevation (above grade)	m	16.8
Top of the containment shield elevation (above grade)	m	~22
Seismic isolation		Yes
Fuel Assembly		
Reference fuel/clad system		U_3Si_2 fuel FeCrAl clad
Fuel pellet		Cylindrical
Solid or annular pellet		Solid
Fuel density	%Th.Dens.	95.5
Pellet O.D.	mm	8.10
Fuel rod O.D. (Clad O.D.)	mm	9.14
Clad thickness	mm	0.406
Fuel rod pitch	mm	12.1
Active fuel height	mm	3,658
Plenum length (lower/upper)	mm	183/274
Fuel lattice		Square, 19x19
Fuel rods per assembly		336
Control rod guide tubes (GT) per assembly		24
Instrumentation tubes (IT) per assembly		1
Fuel Assembly		
Number of fuel assemblies		121
Core equivalent diameter	m	2.867

While fuel is discussed in more detail in the next Chapter, one aspect should be mentioned in this overview. The fuel/clad system is ultimately intended to be composed of high-conductivity and improved (higher) heavy-metal density silicide fuel (U_3Si_2) and low-oxidation rate SiC clad. However, recognizing that fuel qualification is a long end expensive process, the reactor is designed with an option (for lead units) to start operation on current licensed fuel (U_2O/Zirc), transition mid-term to U_3Si_2/FeCrAl, and long-term to U_3Si_2/SiC, as shown in Figure 4.4. Note that even if the reactor is initially started using the current UO_2/Zirc fuel, the overall system safety is enhanced due to improvements in other safety systems. Advanced steel (FeCrAl) cladding reduces the oxidation rate, as compared to Zirc-based materials, however it also introduces a non-trivial reactivity penalty. SiC cladding would further reduce the oxidation rate, while neutronically providing a similar performance as the Zirc-based cladding, but it is possible that its qualification may take longer than FeCrAl qualifications, since steel-based cladding (albeit with different steel type) was historically used in LWRs. This approach provides a flexible and viable path to deployment of improved fuel/clad system, since the initial reactor licensing and operation is not necessarily conditioned by the licensing of novel fuel.

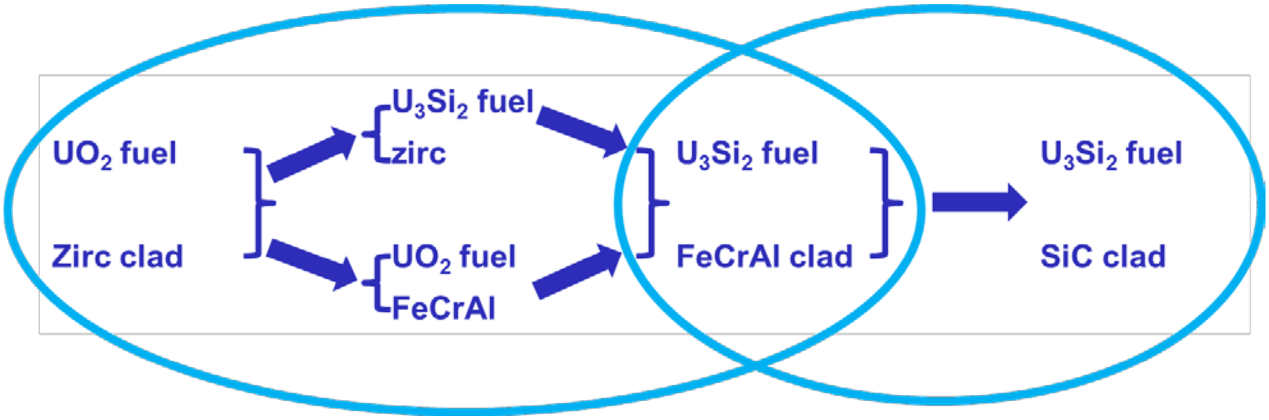


Figure 4.4. Flexible path to fuel with enhanced accident tolerance

Nuclear Island (NI) building is shown in Figure 4.3, with more details presented later. Its footprint is currently listed as 63x69 m; it should be noted that about one third is taken by flash drums and associated pumps. It is expected that future optimization may enable reducing the NI size by 10-15% in each dimension. The whole site layout is presented in Figure 4.5. A topical report details further the nuclear island and site layout.

Nuclear island and site layout details are typically left out at this early stage of pre-conceptual work. It was developed in this project to support a realistic differential economic analysis which demonstrated potential of the I²S-LWR concept to offer a competitive nuclear power option. Overall, the approach was to harmonize safety and economics, i.e., achieve safety via a simple and robust design.

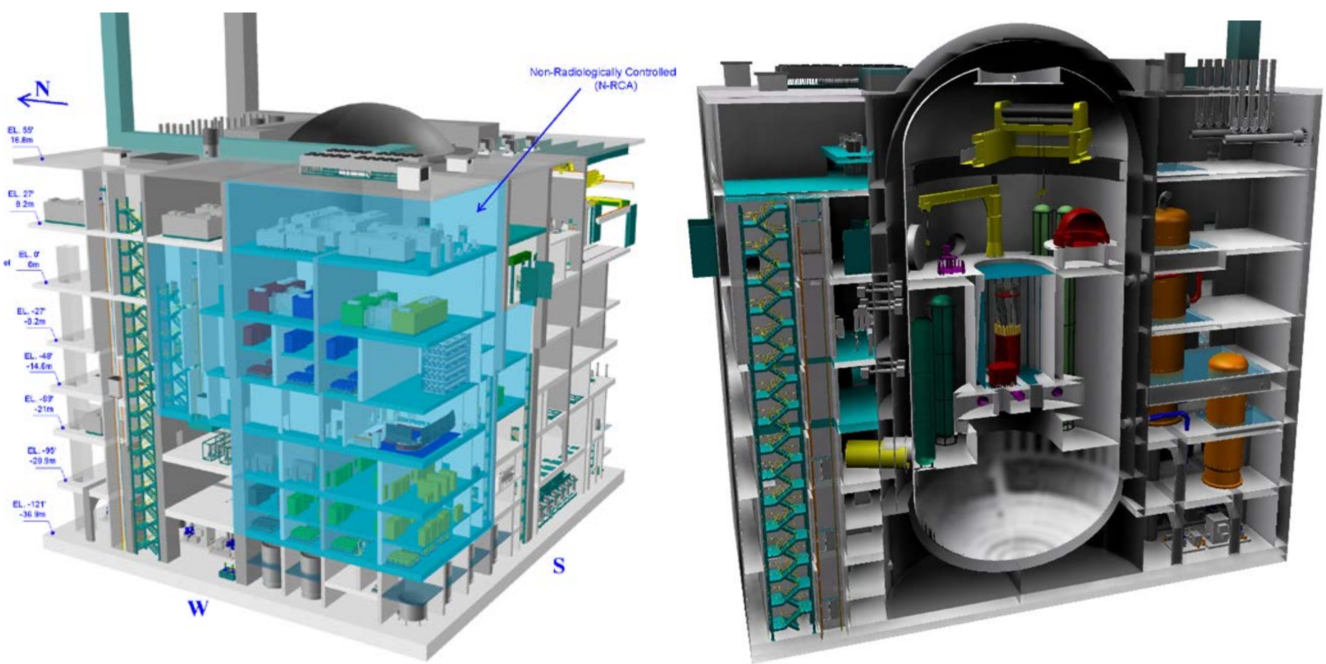


Figure 4.5. Nuclear island – external view and cut-through

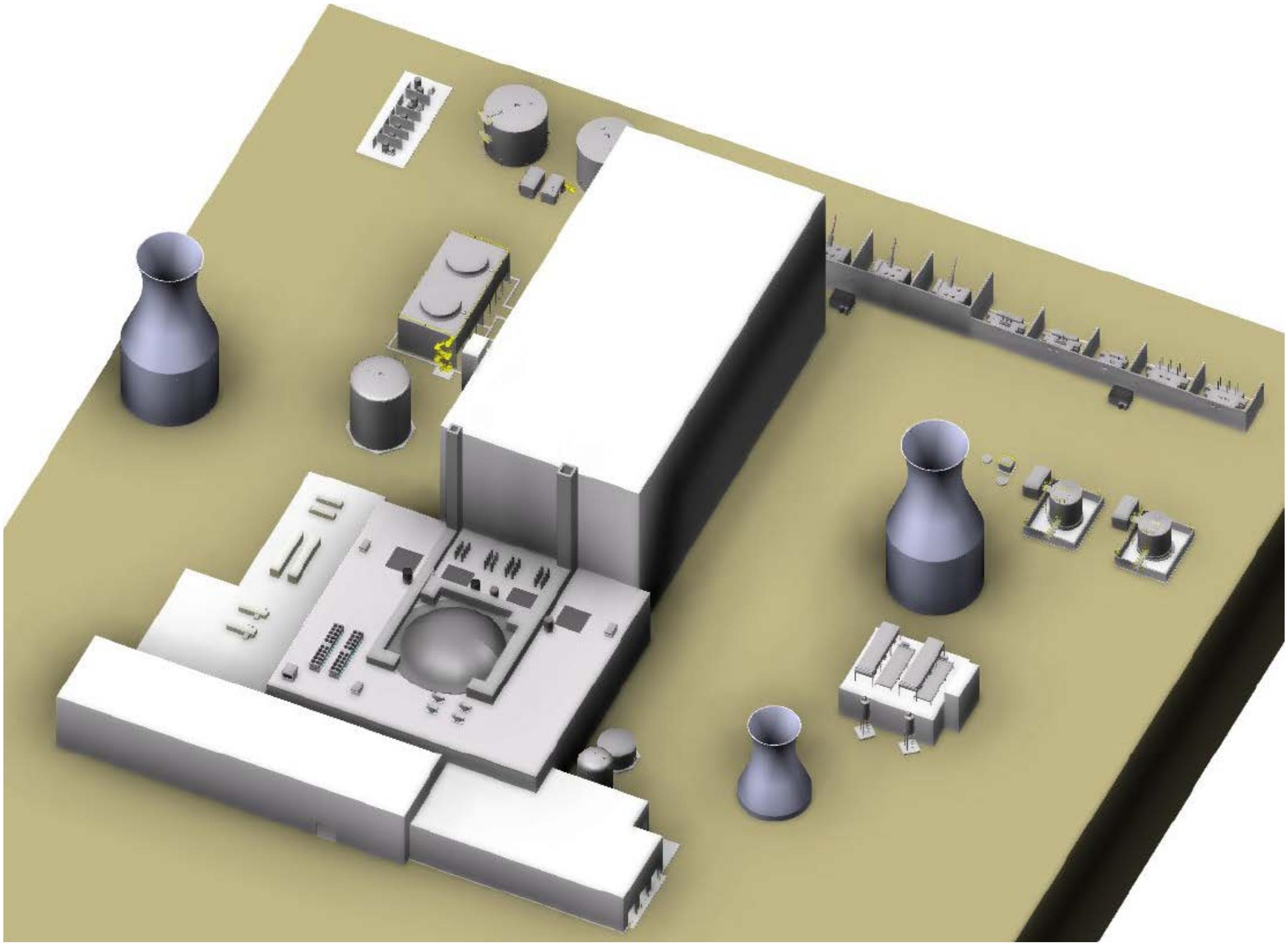


Figure 4.6. General Site Layout – 3D view

The project also served to introduce the students (primarily those in senior design projects) to advanced technologies, including additive manufacturing. During the course of the project a 3D CAD model was developed of the reactor vessel, containment, nuclear island and site layout. 3D models were printed as shown in Figure 4.8 and Figure 4.8. These models proved to be an excellent educational vehicle both for the students that constructed them, as well as for discussion with other students, as they provided an opportunity to better visualize the complexities and challenges of an effective layout and design.

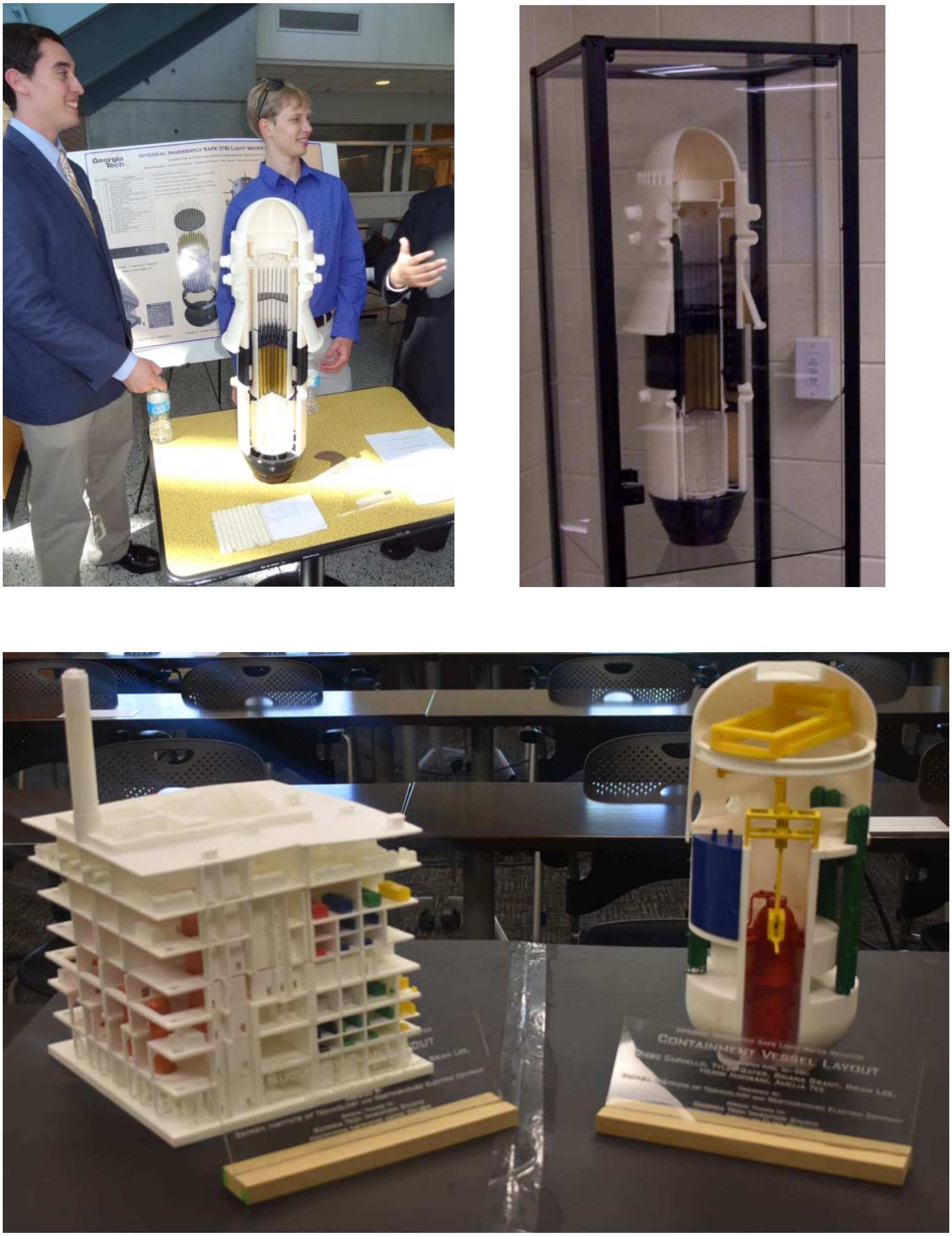


Figure 4.7. 3D models of the reactor vessel, nuclear island and containment

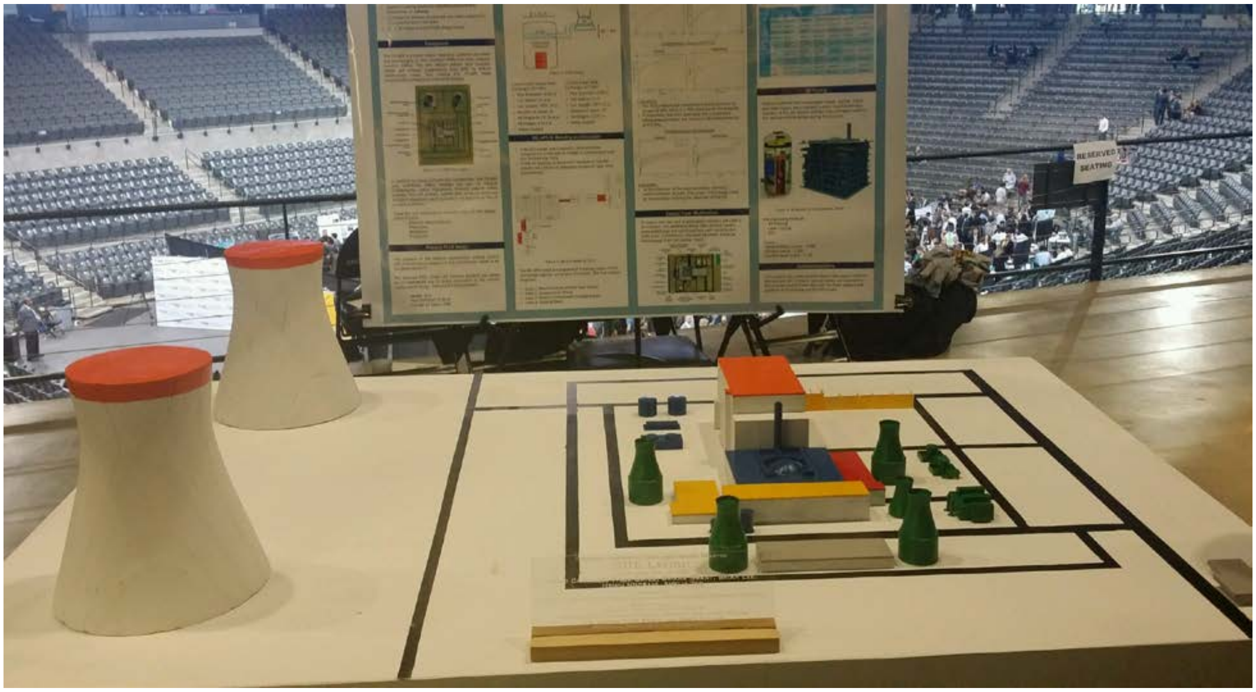


Figure 4.8. 3D models of the plant site

4.4 Project Scope Clarification (In-Scope/Out-of-Scope)

The project accomplished and impressive level of breadth and depth of the design and analyses, in particular considering that it was the initial effort to develop a new concept. However, this does not mean that everything was addressed, or that it could have been performed within this phase of the project and within the available funding. The project had a limited budget, and the scope reflects prioritization, addressing critical items, and specific focus of the proposal. There are a number of tasks/studies that would clearly be beneficial, that need to be performed at some point, which were however not feasible or not justified within this project's. Many of them are good candidates to be included/addressed in the follow-up phase of the project. With encouragement of the External Advisory Board (EAB), a list of in-scope/out-of-scope tasks was prepared, as shown in Table 4.4. This list provided an effective guidance and a mechanism to efficiently address the questions/concerns of the type "why is the project looking or not looking at the following…." The table was periodically reviewed, and in some cases, resources permitting, some out-of-scope tasks were accomplished by leveraging DOE funds e.g. through Senior Design activities. Note that Table 4.4 is not meant to be a detailed and exhaustive list of all possible tasks, but instead is aiming to cover primarily those tasks where a question of "when/why" was most likely to arise and needed to be answered.

Table 4.4. Project In-Scope / Out-of-Scope list of tasks.
(NOTE: By default, anything not specified in In-Scope is Out-of-scope)

Task, action, component of system	**In-Scope**	**Out-of-Scope**	**Comment**
Components			
Primary coolant pumps	Define number, position, and performance requirements Evaluate basic feasibility	Actual design	
Primary heat exchangers	Representative design of a MC-HX module Testing of fundamental correlations for MC HX Basic performance testing based on scaled-down facility	Actual design Operational performance Maintenance	
Control rod drive mechanisms	Review WEC SMR approach and CRDMs (We are reasonably confident that it should work)	Out of scope	Rely on previous studies and activities by WEC SMR and other SMRs
Vessel internals	Concept Be aware of taller vessel	More detailed beyond that	Analogous to large PWRs
Radial reflector	Preliminary geometry, estimate of cooling channels and bypass flow	More detailed evaluation	
Core support		Out of scope	
Reactor vessel	Reasonably detailed conceptual layout		
Containment vessel	Conceptual layout Preliminary vessel-containment performance assessment		
Components testing		Out of scope	
Systems			
Passive decay heat removal system	Concept and conceptual design Basic feasibility assessment		

	Preliminary analyses and evaluation with performance optimization		
Pressurizer and automated depressurization system (ADS)	Concept and conceptual design Basic feasibility assessment Preliminary analyses and evaluation (as SB-LOCA)		
Steam generation system (SGS) including flashing drums and Power conversion system (PCS)	Concept and conceptual design Basic feasibility assessment Preliminary analyses and evaluation with performance optimization		
Turbine building		Out of scope	
Switchyard		Out of scope	
Core design			
Fuel assembly mechanical design	Best guess based on the current PWR robust FA design. Inform by FIV performed on a relative basis.		
Refueling strategy	At least 12 months		
Operating mode	Baseload	Load follow	
Safety Analyses			
Transients	Preliminary analysis of selected representative scenarios		
Accidents	Preliminary analysis of selected representative scenarios		
SAM	Be aware of	Out of scope	
I&C			
Instrumentation	Type of instrumentation, requirements, positioning	Actual design, maintenance	
In-core instrumentation routing	Important issue, not possible to design in detail within the current scope, but address at some level		
Digital I&C	Be aware of	Out of scope	Industry-wide effort

Licensing			
Interaction with NRC	We will probably not reach appropriate level of maturity to engage NRC, but periodically should revisit this option		
Plant licensing issues	Be aware of	Beyond that	
New fuel and cladding licensing	Be aware of	Beyond that	
Operation			
O&M	Be aware of Consider impact for MC-HX since in vessel	Out of scope	
MSHIM	[Perhaps review AP1000 MSHIM and guess whether an analogous approach is feasible for I2S-LWR]	Out of scope	Could be done later, given resources (which are not trivial)
Staffing issues	Be aware of	Out of scope	
Full automation	Be aware of	Out of scope	
Control room		Out of scope	
Refueling	Be aware of taller vessel	Out of scope	
Dry or hybrid cooling	Keep in mind	Out of scope	Separate issue
Cyber security	Be aware of	Out of scope	Industry-wide effort
Construction		Generally out of scope	
Reactor vessel shippability	Very basic assessment		
Cost			
Fuel cost	Estimate of relative difference to current PWR fuel cost		
Capital cost	Estimate of relative difference in NSSS cost to current large loop PWR construction cost	Absolute cost Comparison to SMRs and non-LWR reactors Comparison to other power sources	Sufficient detail cannot be available at this stage. This will be the best available estimate effort

5. Fuel with Enhanced Accident Tolerance – Basic Considerations

5.1 Requirements and candidate materials

To enhance accident tolerance, two main characteristics of the I^{2}S-LWR fuel were envisioned in the proposal:

- Reduced oxidation rate of cladding, to avoid or delay cladding failure;
- Increased thermal conductivity of fuel, to prevent fuel melting.

Additionally, in order to be able to “fit” a 1,000 MWe core in an integral vessel, the core must be more compact, i.e. with a higher volumetric power density, than a typical PWR core. That means faster volumetric burnup rate, and shorter cycle (more frequent refueling) if the same fuel type I used. To counteract the faster volumetric burnup rate, fuel with higher heavy metal (HM) density would be beneficial. Based on these considerations, the proposal identified silicide and nitride fuel, both having a higher thermal conductivity and higher HM density than oxide fuel. Furthermore, two advanced cladding types were identified, advanced FeCrAl steel, and SiC. This is summarized in Table 5.1, Table 5.2 and Table 5.3.

Table 5.1. Fuel and cladding materials considered for the I^{2}S-LWR

	PRIMARY CHOICE	SECONDARY CHOICE	BASELINE
CLADDING MATERIAL	Ferritic stainless steel	Composite SiC	Zircaloy
FUEL MATERIAL	U_3Si_2	^{15}UN	UO_2 (baseline and candidate fuel material)

Table 5.2. Comparison of key properties for the cladding materials

PROPERTY	Zircaloy	ODS-type ferritic steel	Composite SiC
Thermal conductivity (W/m K)	16 (at 370°C, irradiated)	16 (at 370°C, unirradiated, but likely ~irradiated	4-5 (irradiated, ~independent on T)
Melting point (°C)	1825	1500	>2500
High-temperature oxidation rate in steam	X	X/100	X/100
Mechanical properties	Good	Likely better than Zircaloy	Worse than Zircaloy
Neutronic penalty	No	Yes	No
Experience in nuclear applications	Large	Limited to austenitic stainless steels	No
Easy to manufacture	Yes	Yes	Somewhat
		If assumed to behave like austenitic SS, it is much less susceptible than Zircaloy to H_2 uptake and embrittlement	Assuring rod cap sealing is challenging
		If assumed to behave like austenitic SS, it retains tritium much less than Zircaloy	Some properties are anisotropic and manufacture-dependent

Table 5.3. Comparison of key properties for the fuel materials

PROPERTY	UO_2	U_3Si_2	UN
Theoretical/HM densities (g/cm³)	10.98 / 9.68	12.2 / 11.3	14.3 / 13.5
Thermal conductivity, k (W/m K)	5-2 (300-2000°C)	9-20 (300-1200°C)	18-23 (300-1200°C)
Specific heat (J/kg K)	280-440 (300-2000°C)	230-320 (300-1200°C)	220-260 (300-1200°C)
Melting point, T_m (°C)	2840	1665	2760
Thermal margin: $k \times (T_m - 400°C)/100$	50-90	>100	>400
Irradiation-induced swelling	Low	Likely high	Likely high
Reaction with water: negligible?	Yes	Likely yes	No, but may be reduced
Experience in nuclear applications	Large	Some	Some
Easy to manufacture	Yes	Less than UO_2	Much less than UO_2 (it requires ^{15}N enrichment)

The key properties are for cladding and fuel are compared with the baseline materials (UO_2 and Zircaloy). A coloring scheme is used to indicate properties that are definitely better than that for the baseline materials (green), versus those that are definitely worse (orange). Properties for which the difference from the baseline materials is either not significant or not yet well understood are left uncolored.

5.2 Materials database

To be able to consistently and accurately evaluate various options, an extensive literature search was performed, and a database with material properties, for all fuel/clad materials of interest was prepared based on a critical review to identify the most credible values. The database is included as an Appendix to the topical report on Materials.

This critical review also enabled identifying the gaps, as well as the ongoing synergistic DOE programs, and consequently defining a limited but targeted set of experiments to be performed within this project.

5.3 Trade-off studies and fuel/clad system down-selection

Trade-off studies were performed during the first project year aimed to down-select one reference fuel/clad option among the candidates. This is a fairly standard approach, that was applied in several other I²S-LWR design areas. However, while in other areas the down-selection was achieved pretty fast, in the case of fuel/clad, the assessment was that it is prudent to keep several options and a flexible path that accommodates them was developed. The main concern is that fuel qualification and licensing takes time and money, and there is a large uncertainty involved. Therefore, it is desirable to have the option to start the NPP on current, licensed fuel (UO_2/Zirc), and introduce advanced cladding and/or advanced fuel as it becomes qualified and licensed. This flexible incremental deployment path is illustrated in Figure 5.1

Note that SiC cladding is the long-term clad target. This is due to its better properties (from the accident tolerance aspect), combined with lower neutronics penalty. FeCrAl has a non-negligible reactivity penalty, while SiC is not too different from Zircaloy in that respect. Other than oxide fuel, only silicide fuel has been considered as advanced fuel type in this project. This is due to several factors. At the time when the trade-off studies were performed, it seemed that silicide may have a lower oxidation rate, and it was not practical to pursue to many combinations of fuel/clad. Nitride was already pursued by other teams, so we decided to focus on silicide. However, if nitride turns out to provide better fuel performance, it can easily be substituted as well. It has a higher HM density, and a higher thermal conductivity. Therfore, a silicide fuel design that has satisfactory core physics and fuel cycle with silicide, is likely to also work (with some modifications) for nitride (assuming that adequate stability and low oxidation rate of nitride are demonstrated).

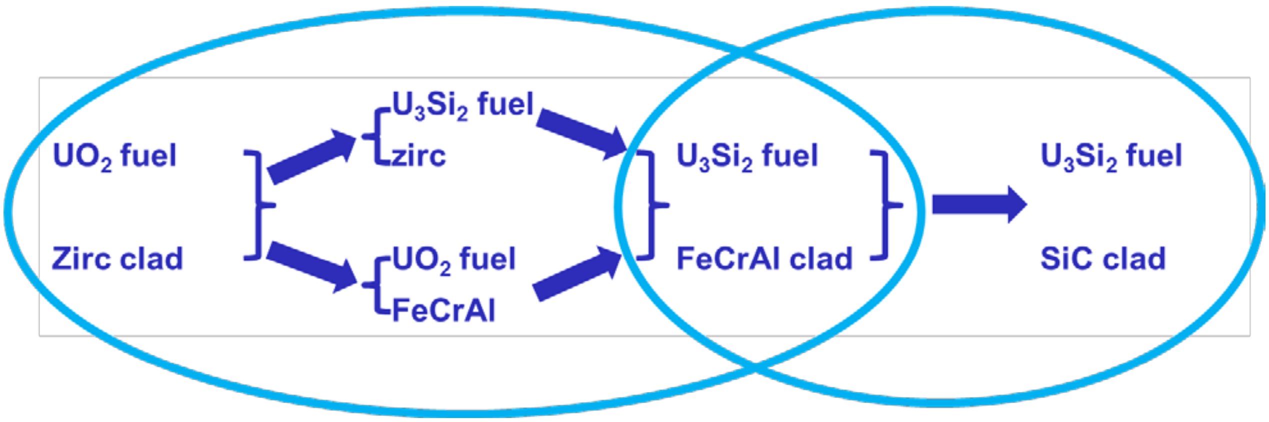


Figure 5.1. Flexible path to fuel with enhanced accident tolerance

5.4 Design challenges related to clad properties

One aspect of the design challenge was driven by the unknowns and uncertainties in properties and performance characteristics of some materials. Considering the new clad candidate materials, FeCrAl properties were sufficiently well known to allow analysis with a high level of confidence. There is a significantly higher uncertainty related to SiC properties and performance in the reactor environment. Therefore, most analyses were performed assuming FeCrAl cladding. FeCrAl has high thermal cross section, resulting in a non-negligible reactivity and fuel cycle cost (FCC) penalty. Thus, results obtained using FeCrAl provide a conservative estimate of economic performance. SiC cladding, when demonstrated, would reduce FCC and improve response in transients and accidents. Thus, the satisfactory outcome obtained with FeCrAl cladding suggests that performance with SiC cladding will also be satisfactory, and in fact better.

5.5 Design challenges related to silicide swelling

Another challenge was related to the uncertainty in the silicide fuel swelling. It is generally believed that it may be higher than that for oxide fuel, but it was not known by how much. This is a critical performance parameter that may significantly impact fuel design. Large swelling would require use of annular fuel and a larger gap, which would eventually negate the benefits of silicide fuel.

Therefore, during the course of the projects, silicide swelling was investigated, and the developed models (detailed subsequently) suggested that swelling would remain limited for the burnup range of interest.

This enabled switching to the solid fuel pellet design and reducing the initially assumed large pellet-to-clad gap. The results are detailed in two journals papers provided in Appendix to the topical report "Materials". Both papers examined fission gas bubble formation and U_3Si_2 fuel swelling in bulk material.

Most of the experimentally available silicide swelling data were obtained at relatively low temperatures in research reactors. Under these conditions, silicide becomes amorphous quickly and remains amorphous up to a fairly high fluence. Study [1] performed by Winter et al. models swelling for amorphous silicide fuel which allows comparison to the experimental data; good agreement is observed as shown in Figure 5.2.

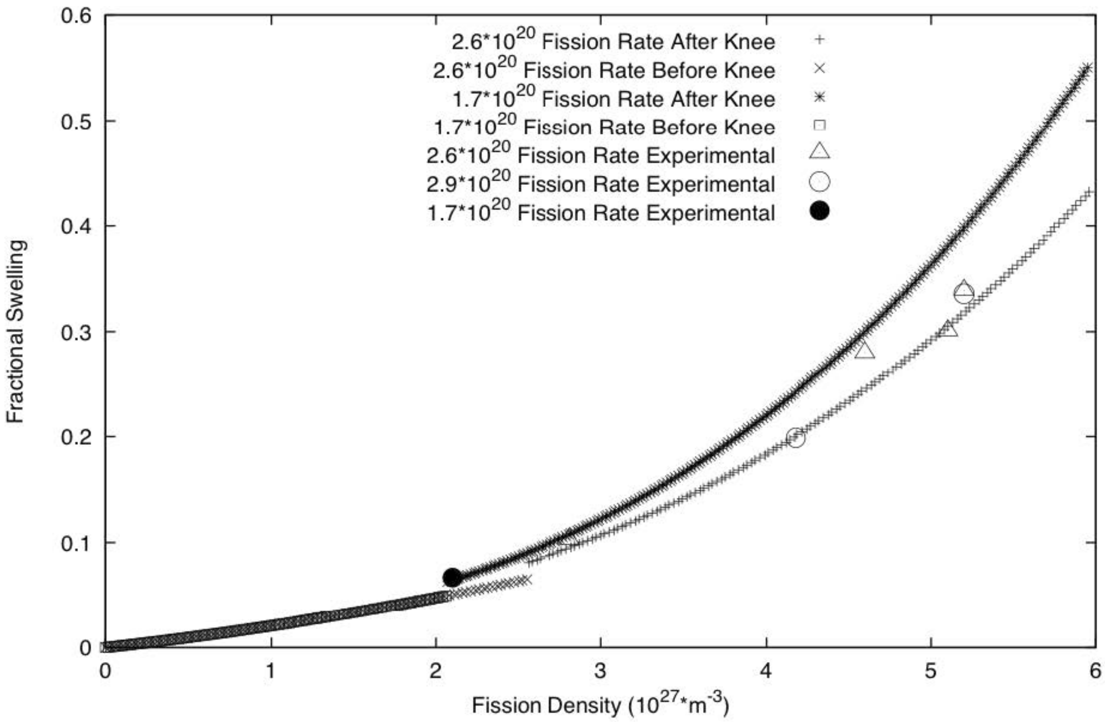


Figure 5.2. Fractional Swelling vs Fission Density for U_3Si_2 for various fission rates compared to experimental values

Swelling before the knee is fairly limited and acceptable for these relatively high fission rates. Depending on the fission rate, the knee would correspond to burnups from 69 GWd/tU to 85 GWd/tU. In any case, the anticipated discharge burnup in I^{2}S-LWR is lower, and swelling would not be a problem. However, the fission rates expected in I^{2}S-LWR are significantly lower. At the fission rates corresponding to the specific power rates in I^{2}S-LWR, the knee point would likely occur during the fuel irradiation, as shown in Figure 5.3. For example, for the specific power of 40 W/gU, the knee point occurs around 17 GWd/tU, and fractional volumetric swelling at fuel discharge would exceed 0.30, which is definitely not acceptable. This model, however, does not assume any fission gas release, which would reduce swelling. Moreover, it assumes, as already discussed, amorphous form, which is not expected at I^{2}S-LWR operating conditions.

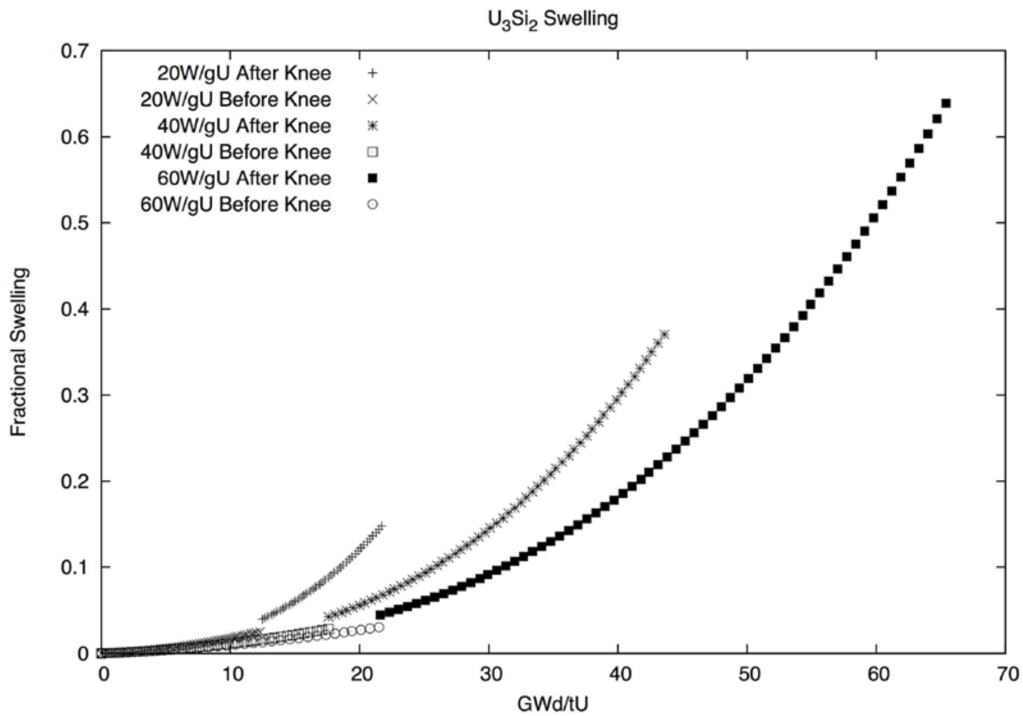


Figure 5.3. Fractional Swelling vs burnup (GWd/tU) for U_3Si_2 and reduced fission rates

The second study [2] performed by Marquez et al. assumes instead crystalline fuel form, representative of I²S-LWR operating conditions, with grain subdivisions impacting fuel swelling The model does not account for any fission gas release, and assumes that recrystallization, if it occurs, will occur over the whole pellet volume, which conservatively overestimates swelling. Since there are no experimental data obtained under similar conditions, by necessity it uses educated guesses for several parameters. Actual values may be different, impacting the swelling estimate. Ongoing silicide irradiation program at ATR may help to improve the values of these parameters. Keeping all these caveats in mind, the results shown in Figure 5-4 indicate recrystallization around 50 GWd/tU, with acceptable swelling before that burnup, and unacceptably high after. This is just about the average predicted discharge burnup for I²S-LWR. However, fission gas release should increase this critical burnup value, and swelling of silicide in I²S-LWR is therefore expected to be acceptable. Experimental work at ATR is expected to confirm or disprove this assessment.

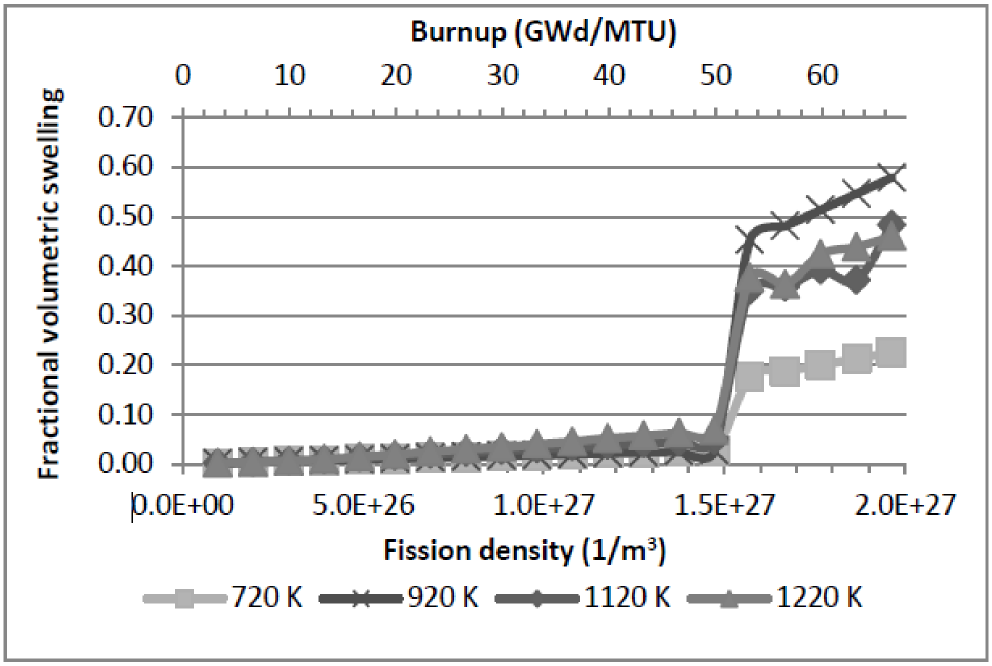


Figure 5-4. Swelling in U_3Si_2 for different temperatures (without fission gas release).

5.5.1 References

[1] T. Winter, C. Deo, "Comparison of fission gas swelling models for amorphous U_3Si_2 and crystalline UO_2 Model for radiation damage-induced grain subdivision and its influence in U_3Si_2 fuel swelling," *Annals of Nuclear Energy*, **100**, 31-41 (2017).

[2] M. Marquez, A. M. Ougouag, and B. Petrovic, "Model for radiation damage-induced grain subdivision and its influence in U_3Si_2 fuel swelling," submitted to *Annals of Nuclear Energy* (2017).

5.6 Experiments

The project had very limited experimental budget. Therefore, it has defined a limited but targeted set of relevant experiments to address some of the key questions related to fuel/cladding.

- High temperature clad oxidation/corrosion experiments (Georgia Tech, Dr. Preet Singh's group)
- Clad/fuel compatibility, i.e., silicide/clad diffusion coupling testing (Georgia Tech, Dr. Chaitanya Deo's group, contributed to experiments at LANL)
- Clad mechanical properties (Dr. Indrajit Charit's group at U. of Idaho)

These experiments are further documented in the topical report on materials.

Additionally, an experimental test facility was built for testing of the MCHX performance (Georgia Tech, Dr. Srinivas Garimella's group); this is further described in the topical report on steam generation system.

6. High Power Density Core Thermal Performance Assessment

6.1 Analysis objective

The designs selected for the I²S-LWR fuel assembly and reactor core have been developed with the objective of generating and safely removing an anticipated thermal power equal to 2850 MWt. Given the target plant thermal efficiency of 35%, this thermal power would result in an electric output of about 1000 MWe, in line with the GW-class plant rating objective. A thermal hydraulic analysis has been conducted, to determine the core operating conditions in terms of core inlet/outlet temperature and coolant flow rate, that would allow the I²S-LWR not only to generate and safely remove this power, but also to provide the Balance of Plant with a heat source at a temperature high enough so that the target efficiency can be achieved.

6.2 Main achievements

An assessment of the thermal hydraulic performance of the I²S-LWR core has been performed, covering steady-state full power operation and a Complete Loss Of Flow Accident. It has been demonstrated that, in order to approach core outlet temperatures similar to those of existing PWRs, and therefore aiming at similar if not higher plant efficiencies, the temperature at the inlet of the I²S-LWR core must be relatively high, around 298°C, which is however within operating experience.

6.3 Approach to analyses

The core thermal performance assessment started with a preliminary investigation which assumed a very low core inlet temperature, T_{in}= 279.4°C, and was performed across a range of core power levels, from the reference value up to about 3100 MWt ([1]). The rationale behind the low T_{in} was the emphasis originally given to the need to limit coolant velocity, and thus grid-to-rod fretting phenomena. This is because a high velocity was anticipated to be required to safely remove the relatively high power density of the I²S-LWR core, and lowering T_{in} was deemed to be an appropriate measure to limit it. At the same time, an understanding of the implications of higher core powers on the operating conditions was needed, which motivated considering power levels above the reference value. This analysis is not repeated here, but is available in quarterly progress reports,

Findings from the initial assessment showed that, with a low T_{in}, coolant velocities in the core ranging between 105% and 115% of the velocity in a typical 4-loop PWR (~4.6 m/s) would be needed to safely remove a thermal power of 2850 MWt, and higher velocities would be needed for higher powers. These percentages become lower, ranging between 95% and 105%, if comparison is made against high-velocity PWRs, where velocity values up to about 5.1 m/s can be seen ([2]). However, in all the cases examined, the low T_{in} also results in a low core outlet temperature, T_{out}, and therefore in a low temperature "seen" by the Balance of Plant fluid in the Primary Heat Exchangers (PHE), which is very detrimental for a key aspect determining the attractiveness of a reactor concept, i.e. its thermodynamic efficiency and therefore economics. For this reason, it was decided to look at higher T_{in} and to focus on the reference core power only, i.e. 2850 MWt, which will be also used in follow-up analyses with the purpose of narrowing down design choices as more information on the plant design and materials will become available. While, on one hand, increasing T_{in} allows higher T_{out} to be obtained, on the other it also implies that higher coolant velocities are needed to satisfy the thermal hydraulic constraints imposed, mainly a maximum void fraction in the subchannels and MDNBR. These velocities are about 30% higher than in most PWRs, and about 20% higher than in high-velocity PWRs. Analysis refinements are undergoing to adopt the most recent results obtained across several areas of I²S-LWR design and assess whether a reduction of this velocity is possible. While it would desirable to do so, experimental results recently obtained elsewhere ([3]) seem to indicate that the cladding material envisioned for the I²S-LWR, i.e. a FeCrAl alloy, has a higher resistance to fretting than Zircaloy.

This chapter of the report is organized as follows:

- Section 6.4 (Analysis method and assumptions) introduces the I²S-LWR power density in the context of the power density of operating plants, and discusses the analysis method
- Section 6.5 (Initial results with T_{in}=279.4°C and 2850 to 3125 MWt core power levels) summarizes some of the results originally obtained with a low T_{in} and multiple core power levels
- Section 6.6 (Recent results for T_{in} >290°C and 2850 MWt core thermal power) summarizes the most recent results, which are focused on the reference I²S-LWR core thermal power of 2850 MWt.

6.4 Analysis method and assumptions

This section is organized as follows:

- Section 6.4.1: Power density comparison with existing plants
- Section 6.4.2: Analysis method
- Section 6.4.3: Code used
- Section 6.4.4: Core geometry
- Section 6.4.5: Assumed steady-state and CLOFA operating conditions
- Section 6.4.6: Constraints used

6.4.1 Power density comparison with existing plants

The I²S-LWR core is a high power density core if compared to existing PWRs. Table 6-1 compares the key power-related parameters, i.e. volumetric power density, average Linear Heat Generation Rate (LHGR) and average heat flux of the I²S-LWR with those of:

- a typical, non-uprated (NU) 4-loop PWR
- a typical uprated (U) 4-loop PWR

This comparison is shown for the reference I²S-LWR power, i.e. 2850 MWt, and for higher powers up to 3125 MWt. It can be seen that, for the reference I²S-LWR design, the difference percentages with respect to the U and NU plants range between about +9 and +16% for the volumetric power density, between -1 and +5% for the LHGR, and between -1 and +9% for the average heat flux. The fact that the LHGR and heat flux difference percentages are lower than those for the power density is the result of the design choices for the I²S-LWR FA, i.e. an increase in the number of fuel rods per FA and a simultaneous reduction of their pitch.

Table 6-1. I²S-LWR power-related parameters and comparison with typical non-uprated PWR ([1]) and uprated PWR ([5], [6])

Parameter	**Unit[a]**	**I²S-LWR values for various thermal powers**			
Core power	MWt	3125	3030	2941	**2850 (ref)**
	MWt/m^3	132.3	128.3	124.5	121.0
	Diff. % NU	26.6	22.8	19.2	15.8
	Diff. % U	19.1	15.5	12.1	8.9
	kW/m	21.0	20.4	19.8	19.2
	Diff. % NU	14.8	11.3	8.1	5.0
	Diff. % U	8.0	4.7	1.7	-1.3
	kW/m^2	731.5	709.3	688.5	668.8
	Diff. % NU	19.3	15.7	12.3	9.1
	Diff. % U	8.0	4.7	1.7	-1.3

[a] Difference percentages are calculated with respect to a typical, 3411 MWt, 0.374" rod OD non-uprated (NU) and a 3626 MWt, 0.360" rod OD uprated (U) PWR. The values for the NU (Watts Bar original design) and U (Vogtle Units 1&2) plants are not necessarily shown in Ref. [1], [5] and [6], but have been calculated using the core thermal power and the core and fuel rod geometry shown in these references.

6.4.2 Analysis method

The core operating conditions searched through this analysis are determined as the conditions that allow the I²S-LWR to remove the desired power while satisfying some key thermal-hydraulic constraints, not only during steady-state full power operation (SS) but also during one of the accidents that is considered a potentially limiting event for the I²S-LWR, i.e. a Complete Loss Of Flow Accident (CLOFA). The reason is that, unlike in loop-type plants, in integral plants the design of the Reactor Coolant Pumps (RCPs) is more constrained, due to space limitation and layout considerations. As a consequence, due to their size and weight, high-head, high-flow, flywheel-provided RCPs of typical loop-plants cannot be incorporated in the I²S-LWR design, and smaller pumps will likely be needed. As a consequence of this, RCP coastdown upon CLOFA is expected to be much faster than for typical, loop-type plants, making CLOFA an important event to consider in the I²S-LWR design due to the important effect that coolant flow has for this event. The methodology used to perform the analysis is discussed in the following subsections. Specifically:

- Section6.4.3: Code used;
- Section 6.4.4 and 6.4.5: Core geometry and assumed operating conditions;
- Section 6.4.6: Thermal hydraulic constraints used

6.4.3 Code used

The thermal-hydraulic analysis of the I²S-LWR high power density core was performed with the Westinghouse version of the VIPRE-01 subchannel code ([8]), referred to as VIPRE-W. The code was used to model 1/8th of the I²S-LWR core, by adopting a subchannel-level description for the central FA (i.e. each rod and subchannel in this FA, which is assumed to be the hot FA, is modeled explicitly) and by decreasing the level of modeling detail as the distance from this FA increases. Specifically, each of the FAs that does not communicate with the central assembly is modeled as a single channel (and single rod), while each of the FAs adjacent to the central one is represented by lumping multiple subchannels into, generally, four larger channels.

6.4.4 Core geometry

Table 6-2 summarizes the geometric parameters of the 121 FAs that form the I²S-LWR core, and compares them to those of existing Westinghouse assembly designs, namely the RFA ([1]) and the OFA ([6]) designs.

Table 6-2. I²S-LWR fuel assembly geometry and comparison with existing designs

Parameter	**I²S-LWR FA design**	**Existing PWR FA design**	
		RFA design ([1])	**OFA design ([6])**
Lattice type	19×19, square	17×17, square	
Fuel type	U_3Si_2 (UO_2 also analyzed)	UO_2	
Cladding material	FeCrAl	Zr-alloy	
Fuel rods/FA	336	264	
Fuel rod OD (mm)	9.14	9.50	9.14
Fuel rod clad thickness (μm)	406	571	
Fuel rod pitch (mm)	12.11	12.60	
Pellet OD (mm)	See Table 5-3	8.19	7.84
Pellet ID (mm)	See Table 5-3	0	
GT/IT per FA[a]	24/1	24/1	
GT and IT OD (mm)	11.05	12.24	12.04
FA pitch (mm)	231	215	

[a] GT: Guide Tubes; IT: Instrumentation Tubes

In fuel rod design, the pellet and cladding geometries must be properly selected so that the stresses resulting from pressure differentials and from pellet-to-clad mechanical interaction (PCMI) do not lead to either cladding failure or departure from a coolable geometry. For the I²S-LWR a detailed fuel rod design has not been performed yet. Instead, a simplified approach has been used, which adopts a 1% End

Of Life (EOL) tensile strain limit on the cladding[1], to protect it from the PCMI resulting from thermal expansion and, especially, from irradiation-induced fuel swelling. In fact, as already discussed in previous quarterly reports, it is not excluded that U_3Si_2 could have a higher irradiation-induced swelling than UO_2, even though a large uncertainty exists on this property. In this analysis, isotropic swelling is assumed, with a conservative EOL peak value equal to 12% dV/V, which compares to about 4-6% for UO_2 at typical PWR peak burnup (4.8% has been assumed for UO_2 in this analysis). Because of this large swelling, for the U_3Si_2 fuel option, in addition to a typical solid pellet an annular pellet provided with an (uncooled) central void is also examined. This geometry would allow the pellet-to-clad gap width to be reduced with respect to that of a solid pellet (since swelling can also occur inward) thus benefitting fuel temperature.

The fuel pellet geometries considered in this study are referred to as I2S-1 through I2S-4, and are summarized in Table 6-3. This table also shows the pellet design for the existing Westinghouse designs mentioned in Table 6-2. In addition to the pellet ID and OD, the table also shows the cross sectional area available for swelling inside the rod, as percentage of the fuel cross sectional area. Contributions from the pellet-cladding gap and, when applicable, central void, are also indicated. The four geometries considered for the I²S-LWR are briefly described as follows:

- I2S-1 is a solid U_3Si_2 pellet satisfying the 1% cladding strain limit;
- I2S-2 is an annular U_3Si_2 pellet with the same total A_{free}/A_{pellet} as of the I2S-1 geometry, and with a contribution coming from the gap equal to that of the RFA and OFA designs, i.e. 4.1%. This geometry does not satisfy the 1% cladding strain limit under the assumption of isotropic swelling, but it does if it is assumed that, once outward swelling causes the pellet surface to reach the cladding, swelling continues but only inward, until the central void is completely filled.
- I2S-3 is an annular U_3Si_2 pellet satisfying the cladding strain limit under any swelling behavior. Specifically, it satisfies the strain criterion even in the case that outward swelling does not stop once the pellet surface reaches the cladding, even if the central void has not been completely filled.
- I2S-4 is a solid UO_2 pellet satisfying the 1% cladding strain limit, assuming a 4.8% EOL swelling for this fuel.

Table 6-3. I²S-LWR fuel pellet geometries (I2S-1 through I2S-4) compared with RFA and OFA designs

Geom ID	Fuel	Pellet-clad gap width (μm)	Pellet diameters (mm)		A_{free}/A_{pellet} (%) (G=gap; V=void; T=tot)		
			OD	ID	G	V	T
Ref. RFA ([1])	UO_2	82.5	8.19	0	4.1	0	4.1
Ref. OFA ([6])	UO_2	78.7	7.84	0	4.1	0	4.1
I2S-1	U_3Si_2	184.1	7.96	0	9.5	0	9.5
I2S-2	U_3Si_2	78.7	8.17	1.84	4.1	5.4	9.5
I2S-3	U_3Si_2	174.0	7.98	2.01	9.5	6.7	16.2
I2S-4	UO_2	95.2	8.14	0	4.7	0	4.7

[1] This tensile strain is defined as $100\times(D_{EOL}-D_{BOL})/D_{BOL}$, where D_{EOL} and D_{BOL} indicate the cladding outer diameter at EOL and Beginning Of Life, respectively. The 1% strain limit is about half of the value that would be used for Zircaloy cladding since steels have generally lower creep and higher Young's modulus compared to Zircaloy, thus resulting in higher stresses for the same strain.

6.4.5 Assumed steady-state and CLOFA operating conditions

Table 6-4 shows the operating conditions assumed in the analysis, and it is followed by a discussion on the rationales used for the selection of some of them.

Table 6-4. I²S-LWR core operating conditions

Parameter	**Steady-state (SS) or CLOFA**	**Value**	**Input(I) or output (O)**
Nominal operating pressure (MPa)	SS	15.51	I
Core power (MWt)	SS	Four cases: 2850 (ref) 2941 3030 3125	I
Enthalpy-rise hot channel factor	SS and CLOFA	1.67	I
Axial peaking factor	SS and CLOFA	1.55, chopped cosine	I
Total hot spot peaking factor	SS and CLOFA	2.59	I
Coolant temp. at core inlet (°C)	SS and CLOFA	Multiple values ≥279.4	I
Average coolant T at the inlet of Primary Heat Exchangers[a] (°C)	SS	Multiple values ≤ 330°C	O
Nominal coolant flow rate	SS	Multiple values	O
Coolant bypass fraction	SS and CLOFA	0.05	I
RCP coastdown coefficient	CLOFA	Multiple values between 4 and 10 seconds, see Figure 5-1	I
Core power profile during CLOFA	CLOFA	See Figure 5-2	I

[a] The temperature at the inlet of the Primary Heat Exchangers is, for the I²S-LWR, the equivalent of the "vessel outlet temperature" in conventional, loop-type plants. This is because the I²S-LWR is an integral reactor, and the primary coolant path is entirely contained within the vessel.

Peaking factors

In this preliminary investigation, the peaking factors used for the I²S-LWR are typical design values used for conventional PWRs. Based on the power distribution determined through the neutronics analysis documented in Ref. [7], these peaking factors are appropriate for design purpose, if not very conservative. Future studies will be performed to determine whether the margin resulting from using these peaking factors is reasonable, or if there are the conditions to reduce them.

Average coolant temperature at core inlet

A survey was conducted among the PWRs operating in the US, with thermal power larger than 3000 MWt. The purpose of this survey was to understand, once the I²S-LWR operating conditions are selected and, for some, calculated, how they compare with operating experience. Table 6-5, which summarizes the key results of this survey together with values for the AP1000® plant[2], shows that:

- the minimum core inlet temperature among operating, 3000 MWt+ PWRs is 281.2 °C, with an even lower value for the AP1000 plant, i.e. 279.4 °C;
- the maximum vessel outlet temperature is 329.3°C.

Due to the core high power density and the relatively fast RCP coastdown expected for the I²S-LWR, it was originally anticipated that, in order to satisfy thermal-hydraulic constraints, this plant would need a relatively high coolant velocity. However, high coolant velocity results in enhanced grid-to-rod fretting, which is the main cause of fuel rod failure in existing PWRs. In the initial part of the assessment, whose results are summarized in Section 6.5, emphasis was given to the need to limit this velocity, and it was therefore decided to fix the core inlet temperatures to the lowest value among the plants surveyed, i.e. 279.4°C ([2]). Subsequent findings, related to a too low core outlet temperature and therefore detrimental impact on plant thermal efficiency, suggested to also look at higher inlet temperatures, as discussed in Section 6.6.

Table 6-5. Survey of US operating PWRs with thermal power above 3000 MWt

	Temperatures (°C)				
	Core inlet T	Core T rise	Average core outlet T	Vessel T rise	Vessel average outlet T
Average value among PWRs	289.6	37.1	327.1	34.8	324.8
Maximum value among PWRs	297.8	41.8	331.6	39.1	329.3
Minimum value among PWRs	281.2	32.8	319.7	31.1	317.2
AP1000®	279.4	45.2	324.6	42.9	322.3

RCP coastdown

Depending on their design, and especially on whether a flywheel is present or not, upon loss of power RCPs can coastdown at different speeds. The parameter most often used to represent this "coastdown speed" is the coastdown coefficient, λ, which is measured in seconds and appears in the coastdown exponential law:

$$\dot{m}(t) = \dot{m}(0)e^{-\frac{t}{\lambda}}$$

[2] AP1000 is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

where $\dot{m}$ is the coolant flow rate and t is the time from the start of the coastdown (s). Figure 6-1 shows the flow rate trends corresponding to values of λ ranging between 3 s (fast coastdown) and 15 s (slow coastdown). This figure also compares these trends to the actual flow coastdown assumed in the CLOFA safety analysis of the AP1000 ([9]) and the Watts Bar ([10]) plants. It can be seen that for typical 4-loop PWRs, such as Watts Bar, the actual coastdown can be approximated with the exponential law presented above if the value assigned to λ is between about 12 and 14 seconds. Details on the I²S-LWR RCP design are not available yet, but values for λ are expected to be between 5 and 7 seconds since higher values, although desirable from a CLOFA standpoint, would likely result in large RCPs, which may be challenging to accommodate due to both high stress levels on the reactor pressure vessel (to which they are attached) and to the need to limit the radius of the containment vessel (which needs to contain them).

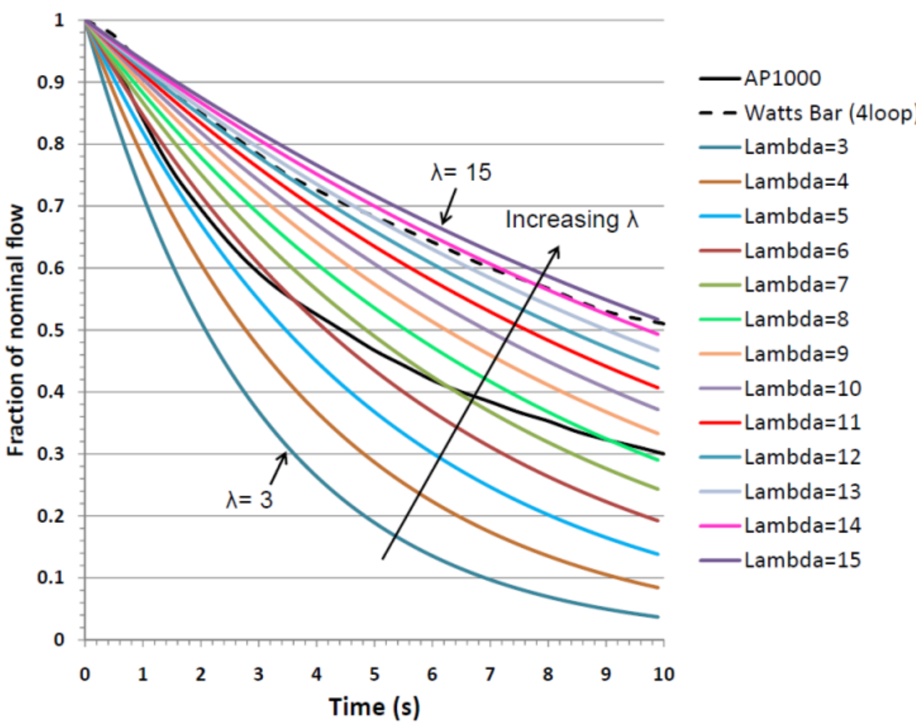


Figure 6-1. Flow coastdown for different values of λ (in seconds), compared with coastdown profiles used for CLOFA analysis of the AP1000 ([9]) and Watts Bar ([10]) plants

Core power profile during CLOFA

In this analysis the nuclear power trend upon CLOFA has been assumed to be similar to that of a typical PWR, and it is shown in Figure 6-2. It must be stressed, however, that this profile will be revised once the design moves forward since plant-specific data such as reactor protection system performance, control rod weight and reactivity worth, guide tubes characteristics and core flow should be accounted for to determine the actual variation of nuclear power during the transient.

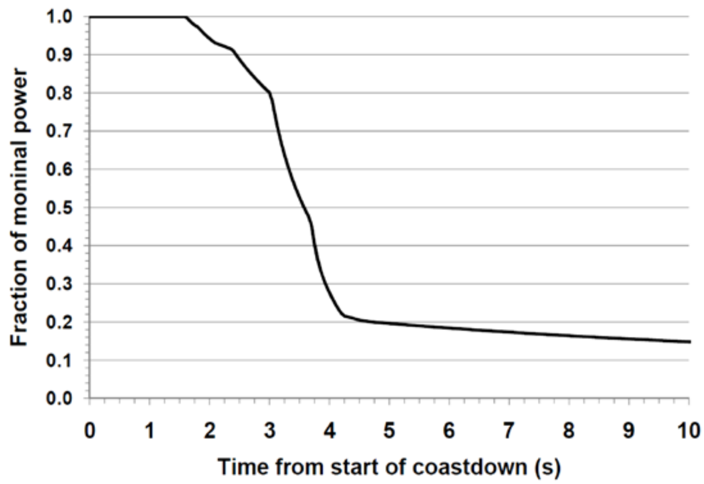


Figure 6-2. Nuclear power vs time assumed for CLOFA analysis

6.4.6 Constraints used

Table 6-6 summarizes the thermal hydraulic constraints used in the analysis, and it is followed by a brief discussion for each. The table indicates whether the constraints are applied to operation during SS or CLOFA, and whether they are hard or soft constraints. A hard constraint must not be exceeded since this will lead to a failure mechanism. Comparison against a soft constraint is instead used as an indication on where the design stands with respect to that limit, and although it is preferable to satisfy the limit, slightly exceeding it may be acceptable. For example, the value selected for the limit on overpower to melting, which represents the linear power increase resulting in fuel melting, is the overpower to melting of an existing UO_2 plant. However, design considerations on, for example, the reactor protection system, may allow this limit to be relaxed. For this reason, while a hypothetical I²S-LWR design with an overpower to melting equal to 1.2 clearly cannot be accepted, a design with a value of 1.6 could.

Table 6-6. Thermal hydraulic constraints used in the analysis

	Steady-state or CLOFA limit	Value	Hard or soft limit
Primary Heat Exchangers (PHE) average inlet temperature	SS	≤ 330°C	Soft
Maximum void fraction in any subchannel	SS	≤ A*	Hard
Minimum Departure from Nucleate Boiling Ratio (MDBR)	Both	≥ 1.60	Hard
Overpower to fuel melting	SS	≥ 1.73	Soft

* A is a Westinghouse proprietary value below 0.3

SS PHE average inlet temperature

In existing PWRs, the SS vessel outlet temperature is limited by the maximum temperature at which the Steam Generator (SG) tubes can safely operate, and it is maintained as close as possible to this temperature in order to benefit plant efficiency. This maximum temperature, for typical SG materials such as Inconel 600 and Inconel 690, is about 620-625 F (326.7-329.4°C). The PWR survey shown in Table 5-3 indicates that the maximum value for operating PWRs is in fact, at the exit of the vessel, 329.3°C. Even though 1) the I²S-LWR PHE are significantly different from shell-and-tube SGs used in typical PWR plants, i.e. of compact, liquid-to-liquid, microchannel-type and 2) their material will likely be different, in this analysis the same temperature as for existing PWRs, rounded up to 330°C, is used, but as a soft constraint. The possibility to use it as a soft constraint is also justified by the fact that the Utility Requirement Document ([12]) suggests plant designers to limit this temperature below 600 F (315 °C). In spite of this, the vast majority of operating PWRs significantly exceed this temperature, since they aim at maximizing efficiency and other measures are in place, or additional analyses have been performed, to guarantee safe operation of the SGs.

It must be noticed that the temperature at the inlet of the PHE results from the core outlet temperature after the effective coolant flow mixes with the bypass flow. Also, unlike the core inlet temperature which is an input parameter and it is maintained fixed in this analysis, the core outlet temperature results from the core power and flow rate used, which are both varied throughout the analysis.

SS maximum void fraction in any subchannel

The coolant flowing through subchannels is allowed to boil, but not to exceed a certain void fraction in order to limit crud formation. At this stage of the I²S-LWR concept development, this void fraction limit is set to a Westinghouse proprietary value, which is smaller than 0.3 and it is used in the design of several Westinghouse fuel products. While this value is appropriate for Zr-based claddings, it is deemed conservative for FeCrAl, due to the improved corrosion performance of this material.

Minimum Departure from Nucleate Boiling Ratio

The MDNBR is monitored to prevent cladding failure due to DNB. This ratio is computed using the Westinghouse WRB-2 correlation ([11]), which is considered adequate for this analysis because of its rather wide FA design applicability range. This range includes the Westinghouse OFA design which, as shown in Table 5-2, has the same fuel rod OD as that of the I²S-LWR assembly design. The MDNBR safety limit, i.e. 1.60, was obtained by applying additive margins, chosen among typical values for conventional PWRs, to the WRB-2 95/95 limit[3] (i.e. 1.17, [2]). These margins are:

- 12% margin to account for both uncertainties in operating parameters (e.g. flow, power, temperature, peaking factors) and phenomena that may occur during operation and that negatively impact DNB (e.g. rod bow);
- 15% margin per Utility Requirement Document recommendation ([12]).

[3] Maintaining the calculated MDNBR above the 95/95 limit implies that there is at least a 95% probability that DNB will not occur, at a 95% confidence level.

The value obtained using these margins, i.e. 1.56, was conservatively increased to 1.60 to account for differences, of the I²S-LWR with respect to typical PWRs, that may negatively impact either DNB directly or the confidence with which plant parameters are known[4].

Overpower to fuel melting

The overpower to fuel melting is defined as the linear power increase that, starting from SS, would result in fuel melting at the core hot spot location. This parameter is calculated by artificially increasing the hot rod linear power until the peak fuel temperature reaches the melting point, which is 1665°C and 2850°C for U_3Si_2 and UO_2, respectively. The limit chosen, i.e. 1.73, is the overpower to fuel melting computed for a reference UO_2 design, and it is used as soft constraint since, as mentioned above, design considerations such as enhancements in core protection system performance, but also elimination of certain accidents, e.g. Rod Ejection Accident, allow for some flexibility in the way the reactor performance compare against this value.

6.5 Initial results with T_{in}=279.4°C and 2850 to 3125 MWt core power levels

Figure 6-3. I²S-LWR coolant velocity at core inlet, as a function of coastdown coefficient and core thermal power, for T_{in}=279.4 °C shows the nominal coolant velocity, as a function of the RCP coastdown coefficient that would ensure the SS and CLOFA thermal hydraulic constraints summarized in Table 6-6 to be met, for a constant inlet temperature equal to 279.4°C and four power levels between 2850 MWt (reference) and 3125 MWt. As a means of comparison, the same figure shows the value for the same parameter but in a typical 4-loop PWR ([4]), and Figure 6-4. Coolant velocity ratio (I²S-LWR/4loop) as a function of coastdown coefficient and core thermal power, for T_{in}=279.4 °C shows the velocity ratio with respect to this velocity. It can be noticed that:

- as expected, the required coolant velocity increases with the core thermal power;
- for a given power, the coolant velocity decreases with the coastdown coefficient, until a certain λ (7-8 s) is reached, after which it remains constant. This is because for slow coastdown RCPs (i.e. high λ) the CLOFA MDNBR constraint is less limiting than the SS void fraction constraint, thus making these high λ scenarios SS limited, not CLOFA limited;
- the coolant velocity for the I²S-LWR is between slightly higher and significantly higher than in a typical 4-loop PWR. However, for the reference power level of 2850 MWt, it can be maintained within 10% of that value if RCPs with a coastdown coefficient above about 5.7 s can be procured and incorporated in the I²S-LWR design.

[4] For example, flow measurement uncertainty is expected to be larger in the I²S-LWR than in a conventional loop-type plant due to the challenge in measuring primary flow rate in absence of "confined" cold/hot legs.

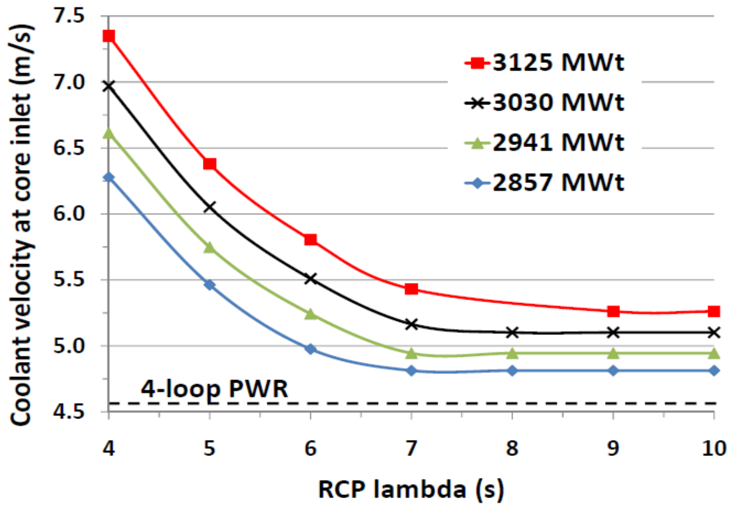


Figure 6-3. I²S-LWR coolant velocity at core inlet, as a function of coastdown coefficient and core thermal power, for T_{in}=279.4 °C

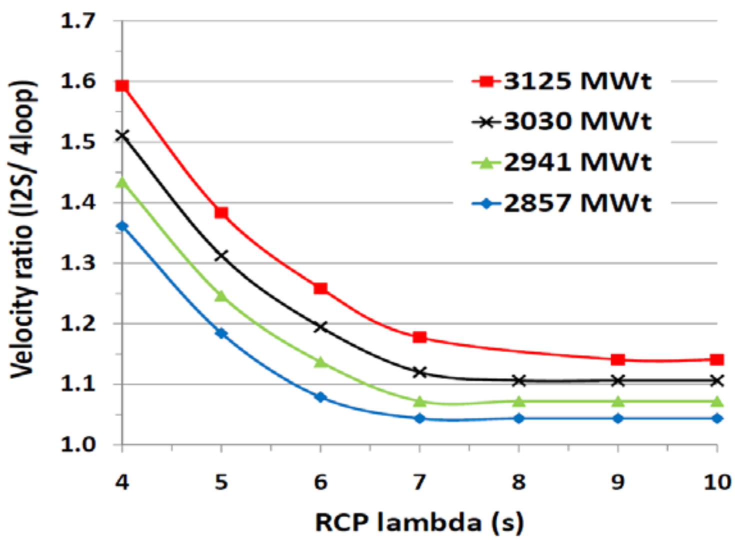


Figure 6-4. Coolant velocity ratio (I²S-LWR/4loop) as a function of coastdown coefficient and core thermal power, for T_{in}=279.4 °C

The primary coolant temperature at the PHE inlet was calculated from the core-average outlet temperature accounting for core bypass flow, and it is shown in Figure 6-5. It can be noticed that, as mentioned, for all the cases analyzed this temperature is quite low, well below not only the maximum allowed value reported in Table 5-6, i.e. 330°C, but especially below the average value of the vessel exit temperatures among the surveyed PWRs, as indicated in Table 6-5, i.e. 324.8°C. Because of the need to have this temperature as high as possible, it was decided to increase the core inlet temperature above the 279.4°C value, aware that coolant velocity would be penalized, i.e. increased. This is discussed in the next section.

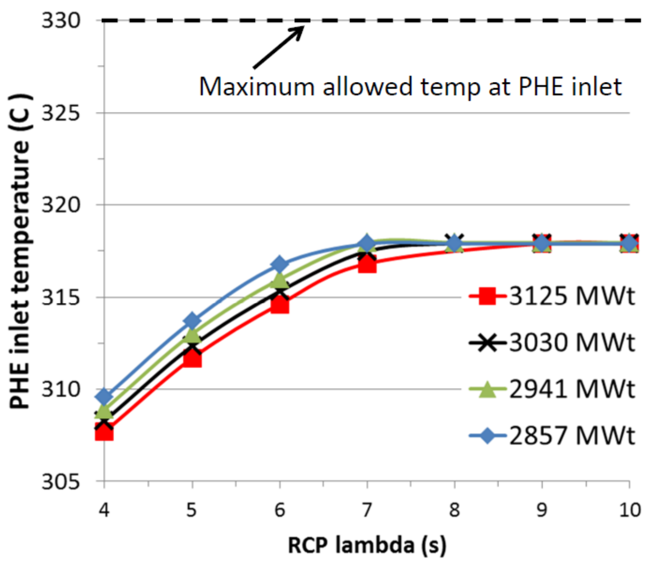


Figure 6-5. PHE primary coolant inlet temperature, as a function of coastdown coefficient and core thermal power, for T_{in}=279.4 °C

6.6 Results for T_{in} >290°C and 2850 MWt core thermal power

As mentioned, in order to achieve an efficiency as high as possible, efforts have been made to increase the core outlet temperature above the low values obtained in the earlier assessment, and shown in Figure 6-5, while still satisfying the thermal hydraulic constraints listed in Table 6-6. To obtain this objective, the core inlet temperature was increased well above the value originally assumed, i.e. 279.4°C, and varied between 290°C and 298°C. For each temperature, and with the core thermal power fixed to the reference 2850 MWt value, the analysis procedure discussed in Section 6.4 was repeated, with the objective of determining the coolant velocity needed to meet the thermal hydraulic constraints, and subsequently calculating the core outlet temperature corresponding to each T_{in}-coolant velocity combination.

Key results of this analysis are presented in Section 6.6.1 (Coolant velocity and PHE inlet temperature) and Section 6.6.2 (Fuel temperature).

6.6.1 Coolant velocity and PHE inlet temperature

Figure 6-6 through Figure 6-8 are the equivalent of Figure 6-3 through Figure 6-5 presented in Section 6.5. Specifically, Figure 6-6 shows the coolant velocity needed for each λ-T_{in} combination, while Figure 6-7 shows the ratio between this velocity and that of the reference 4-loop PWR. By comparing it with Figure 6-4, the same trends can be noticed, but the velocity ratios are higher, as expected because of the hotter coolant conditions at the inlet. Specifically, while with a 279.4°C inlet temperature coolant velocities within 110% of the reference value were possible, with the higher inlet temperatures and assuming λ ≥6.5 s the minimum velocity is about 115% of the reference value (corresponding to the lowest T_{in}), and it increases to about 133% for the highest T_{in} analyzed. However, consistent with the objective tackled through this analysis, these operating conditions also include higher outlet temperatures, as shown in Figure 6-8, and therefore higher plant efficiencies and better economics. In particular, with Tin=298°C, a PHE inlet temperature of about 327°C can be achieved, which is above the average of the vessel exit temperatures among the surveyed PWRs (324.8°C) and slightly below the maximum vessel outlet temperature of the same plants (329.3°C) (see Table 6-5).

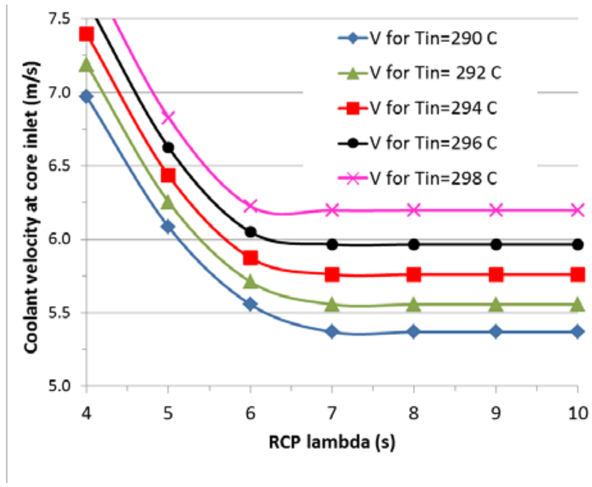


Figure 6-6. Coolant velocity as a function of coastdown coefficient and core inlet temperature, for 2850 MWt thermal power

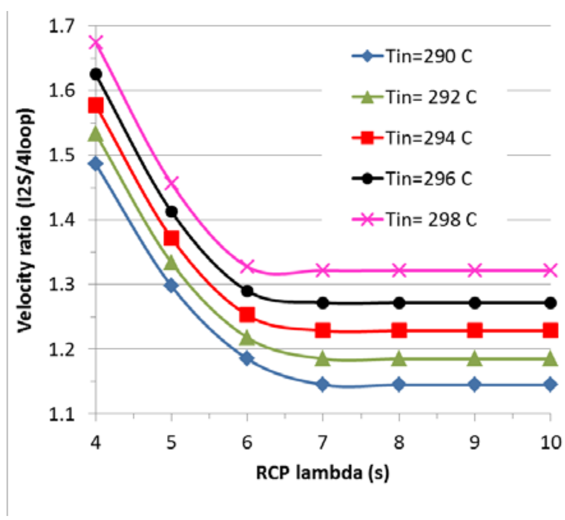


Figure 6-7. Coolant velocity ratio (I²S-LWR/4loop) as a function of coastdown coefficient and core inlet temperature, for 2850 MWt thermal power

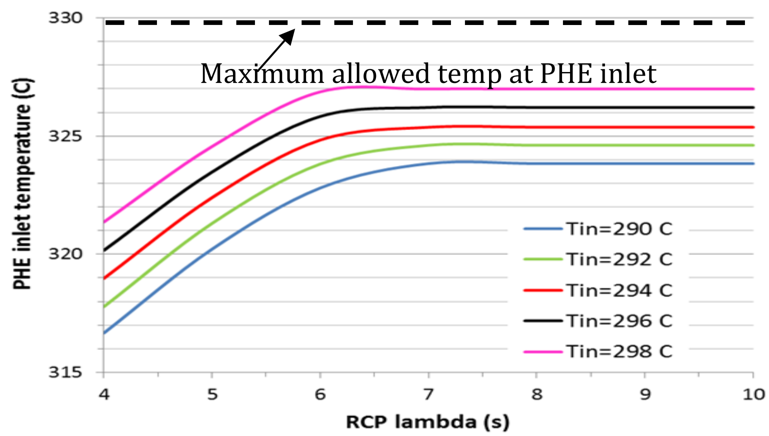


Figure 6-8. PHE primary coolant inlet temperature, as a function of coastdown coefficient and core inlet temperature, for 2850 MWt thermal power

6.6.2 Safety margin to fuel melt in overpower transients

Figure 6-9 shows the overpower to melting, using the reference thermal power of 2850 MWt, for different I²S-LWR fuel options in terms of fuel type (U_3Si_2 and UO_2) and pellet geometry (w/ and w/o void, thin or wide pellet-cladding gap). Details on these fuel options are discussed in Section 6.4.4 and summarized in Table 6-3. In obtaining this figure, the most recent data for U_3Si_2 thermal conductivity have been used. These data have been recently measured experimentally ([13]) and revealed this fuel to be even more conductive than originally thought. The figure, which also contains the 1.73 soft limit used in this analysis (see Table 6-6), shows that the safety margin to melting for U_3Si_2-fueled cores is between equal and larger than for UO_2. In particular, for the U_3Si_2 geometry referred to as "Geom 2" (pellet with a central void and a pellet-cladding gap similar to that of existing UO_2 designs) the safety margin is more than 2.3 times larger than for a typical PWR. Also, the margin for an UO_2-fueled I²S-LWR is approximately equal to that of a typical PWR.

Given the uncertainty on U_3Si_2 swelling, Figure 6-9 clearly shows that the I²S-LWR will have promising performance from the margin to fuel melting standpoint. The reason for this is the 3-6 times higher thermal conductivity of U_3Si_2 relative to UO_2, which is sufficient to compensate for its lower melting point (1665°C vs 2850°C for UO_2). In fact, the temperature increase resulting from the same increase in linear power is between about 2 and 4 times higher for UO_2 than for U_3Si_2, as shown in Figure 6-10. In this figure, the derivative of the peak fuel temperature with respect to linear power is plotted on the left y-axis, while the right y-axis shows the UO_2/U_3Si_2 ratio between the two derivatives.

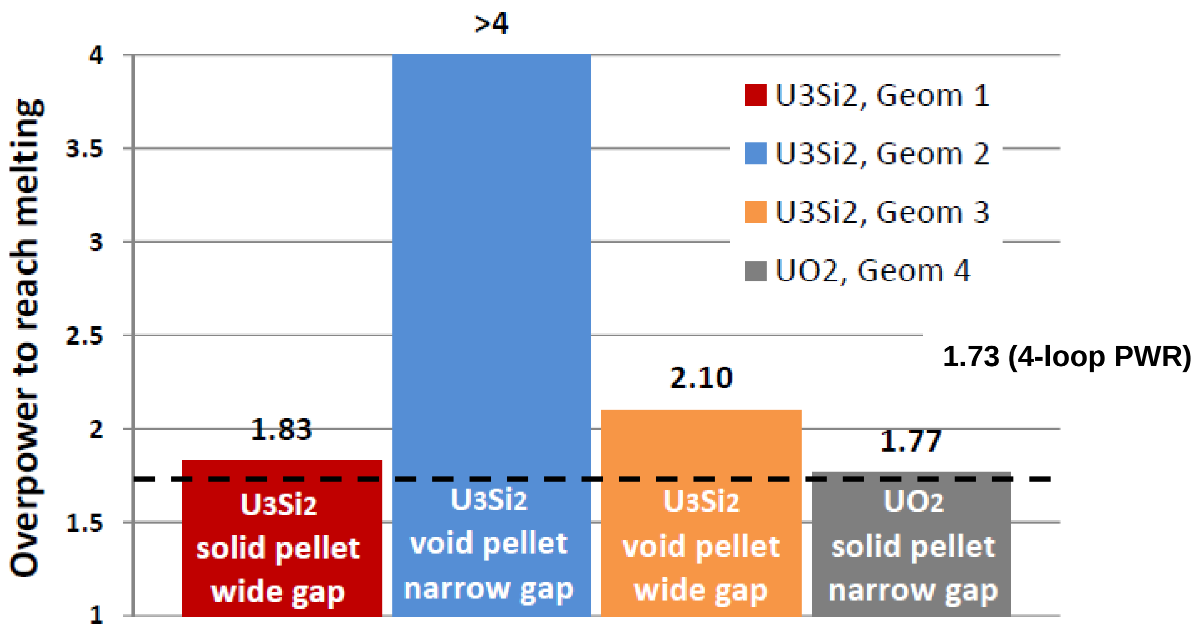


Figure 6-9. Overpower to melting for a 2850 MWt I²S-LWR core, for the different options of fuel type (U_3Si_2 and UO_2) and pellet designs (as summarized in Table 5-3)

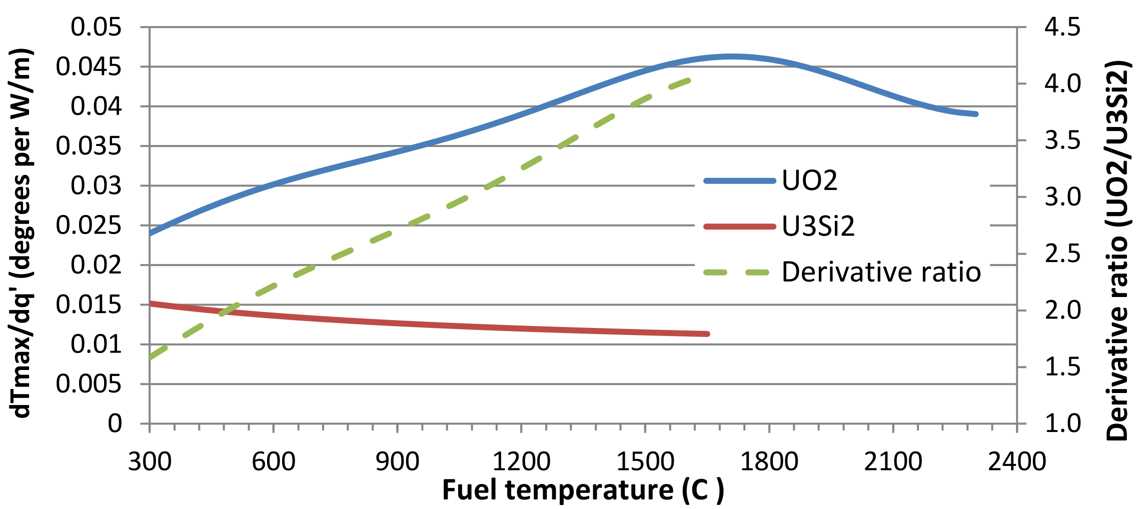


Figure 6-10. Derivative of peak fuel temperature with respect to linear power for constant geometry, gap conductance, cladding properties and coolant heat transfer coefficient

6.7 HPD analyses with higher allowed void fraction due to FeCrAl cladding

Previous work showed that, using a 298°C core inlet temperature and under typical thermal-hydraulic constraints used in the analysis of LWRs ("old" values in Table 6-6), the maximum achievable temperature for the primary coolant entering the Primary Heat Exchangers (PHE) is about 327°C. Although high, this temperature is about 2°C below the maximum hot leg temperature among the US

PWRs, which had been surveyed to identify the temperature operating range for coolant temperature across existing plants. Because of the dependence of the plant thermodynamic efficiency on this temperature, and the importance of such efficiency for achieving the optimal electric power output target of 1000 MWe, further analysis was performed to investigate the effect of relaxing one of the constraints used, i.e. the maximum subchannel void fraction during steady-state, on the maximum achievable temperature at the PHE inlet. The relaxation of this constraint, from a Westinghouse proprietary value lower than 0.3 to a new value equal to 0.3 (see Table 6.7), is well justified for the primary I²S-LWR core design, which employs highly corrosion-resistant FeCrAl cladding. In fact, the main reason why for conventional Zircaloy-clad PWRs the subchannel void fraction is maintained below the "A" value indicated in Table 6.7 is to limit CRUD formation, which results from the coexistence of multiple phenomena one of which is cladding corrosion. Given the superior corrosion resistance of FeCrAl with respect to Zircaloy, experimentally demonstrated both within and outside of the I²S-LWR project, relaxation of this constraint is therefore justified. The new limit, i.e. 0.3, is simply the upper bound of the void fraction validity range of the CHF correlation used in the analysis, i.e. the WRB-2 correlation ([11]).

Table 6.7. Thermal hydraulic constraints used in the analysis

	Steady-state or CLOFA limit	Value	
		Old	New
Primary Heat Exchangers (PHE) average inlet temperature	SS	≤ 330°C	Same
Maximum void fraction in any subchannel	SS	≤ A*	≤0.3
Minimum Departure from Nucleate Boiling Ratio (MDBR)	Both	≥ 1.60	Same
Overpower to fuel melting	SS	≥ 1.73	Same

* A is a Westinghouse proprietary value below 0.3

The analysis method adopted is the same used for previous calculations, and consists of modeling, using the VIPRE-W thermal hydraulic code, the I²S-LWR core during both steady-state operation and a Complete Loss Of Flow Accident (CLOFA). For the latter scenario, multiple cases are modeled to account for different reactor coolant pump (RCP) coastdown performance, which is expressed by means of the coastdown coefficient λ, which is measured in seconds and appears in the coastdown exponential law:

$$\dot{m}(t) = \dot{m}(0)e^{-\frac{t}{\lambda}}$$

where $\dot{m}$ is the coolant flow rate and t is the time from the start of the coastdown (s). A high λ value indicates a slow coastdown, which is preferable during CLOFA. This coefficient has been varied from 4 to 10 seconds, and it is expected that RCPs for the I²S-LWR can reach 6-7 seconds (vs 12-14 sec for large RCPs used in loop-type PWRs). For steady-state operation, and for the various CLOFA scenarios (one for each λ value), the core inlet flow has been iteratively varied in search for the minimum flow that ensures the constraints in Table 6-6 to be met, using a fixed core thermal power of 2850 MWt and a fixed inlet temperature equal to 298°C. Once this flow is found, the PHE inlet temperature and the coolant velocity can be readily obtained. Results are shown in Figure 6-11 and Figure 6-12, for three cases:

- "old" case: maximum subchannel void fraction limited below A% (see Table 6-6), with and maximum temperature at the PHE inlet limited below 330C. This is representative of the achievable performance of I²S-LWR when Zircaloy is used as cladding material.
- "new" case with relaxed limit on the maximum subchannel void fraction (30%), and PHE inlet temperature still limited to 330C. This is representative of the achievable performance of I²S-LWR when FeCrAl is used as cladding material.
- Additional "new" case, with relaxed limit on the maximum subchannel void fraction (30%), and PHE inlet temperature no longer constrained to be below 330C. This is to show the p1otential for a further enhancement in performance of I²S-LWR when FeCrAl is used, in the hypothetical case that the 330C limit[5] on PHE inlet temperature could be relaxed.

Figure 6-11 shows the achievable PHE inlet temperature, in the three cases mentioned above. As already discussed in the previous analysis, the general trend is an increase of the temperature with increasing λ, due to the fact that, for CLOFA scenarios, higher λ values (slower coastdown) allow the MDNBR limit to be satisfied with a lower initial (steady-state) flow. However, such dependence on λ disappears at high values of λ, due to the fact that with very slow coastdown the MDNBR during CLOFA stops being the limiting constraint, and is "replaced" as limiting constraint by the subchannel void fraction during steady-state (which is not dependent on RCP coastdown characteristics).

It can be seen that, with respect to the Zircaloy-clad design that can reach a maximum value of about 328 C, the void fraction relaxation allows the FeCrAl-clad design to achieve 330 C, even with the relatively low λ values (6-7 sec) expected to be feasible with the I²S-LWR RCPs. If no limit was imposed on the maximum PHE inlet temperature, a further increase to between 330.5 and 332.5 C would be possible, depending on the achievable RCP λ value.

[5] In existing PWRs, the SS vessel outlet temperature is limited by the maximum temperature at which the Steam Generator (SG) tubes can safely operate, and it is maintained as close as possible to this temperature in order to benefit plant efficiency. This maximum temperature, for typical SG materials such as Inconel 600 and Inconel 690, is about 620-625 F (326.7-329.4°C). The survey conducted on operating US PWRs indicates that the maximum temperature at the exit of the vessel is 329.3°C. Even though 1) the I²S-LWR PHE are significantly different from shell-and-tube SGs used in typical PWR plants, i.e. of compact, liquid-to-liquid, microchannel-type and 2) their material will likely be different, in this analysis the same temperature as for existing PWRs, rounded up to 330°C, is used, but as a soft constraint. The possibility to use it as a soft constraint is also justified by the fact that the Utility Requirement Document ([12]) suggests plant designers to limit this temperature below 600 F (315 °C). In spite of this, the vast majority of operating PWRs significantly exceed this temperature, since they aim at maximizing efficiency and other measures are in place, or additional analyses have been performed, to guarantee safe operation of the SGs.

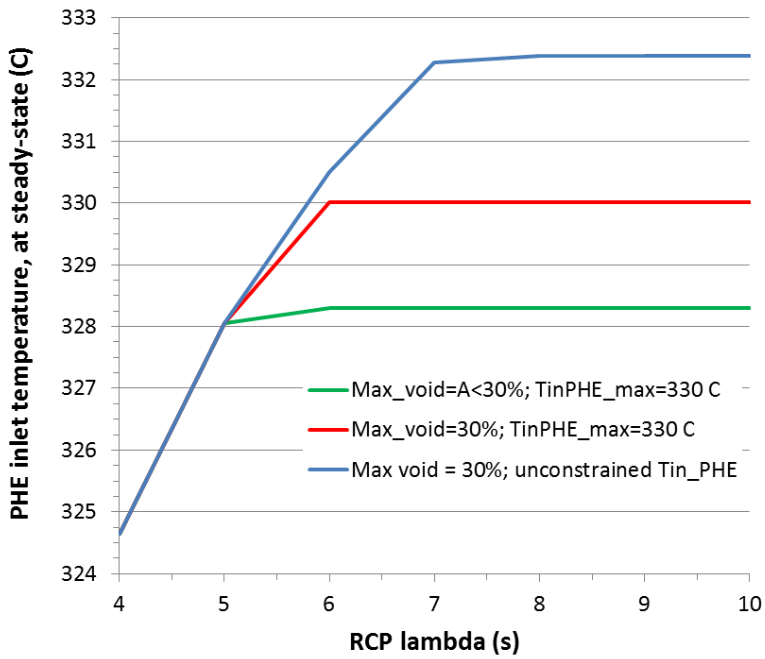


Figure 6-11. PHE primary coolant inlet temperature, as a function of RCP coastdown coefficient, for different cases of maximum subchannel void fraction and limit on PHE inlet temperature (2850 MWt thermal power)

Figure 6-12 shows the coolant velocity in the core, corresponding to the various cases in Figure 6-11, as well as the ratio with respect to the velocity in a reference 4-loop PWR. It can be seen that while the coolant velocities corresponding to the constraints applicable to the Zircaloy-clad design, in the 6-7 sec λ range, are about 30% higher than in the reference plant, with the relaxed constraint applicable to the FeCrAl-clad design this percentage decreases to about 22%. A further reduction, to between 20% and 13%, would be possible if also the limit on maximum PHE inlet temperature was relaxed, for λ values between 6 and 7 seconds, respectively.

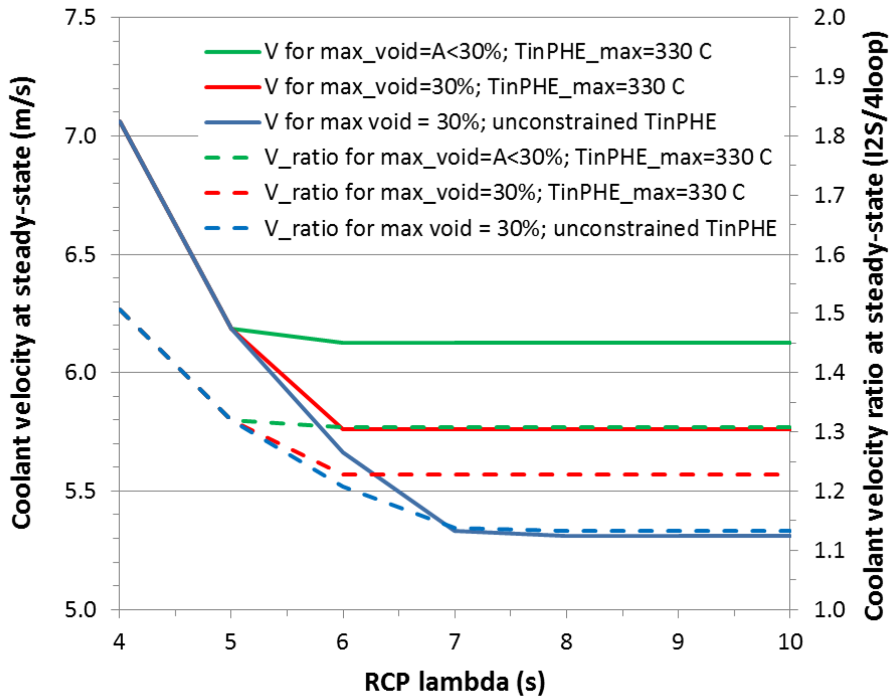


Figure 6-12. Steady-state coolant velocity in the core, and velocity ratio with respect to a 4-loop PWR, as a function of RCP coastdown coefficient, for different cases of maximum subchannel void fraction and limit on PHE inlet temperature (2850 MWt thermal power)

The main conclusion from this assessment is the possibility, with FeCrAl cladding, to achieve primary coolant temperatures, at the inlet of the PHE, higher than those determined in the past, when thermal-hydraulic constraints specific to Zircaloy cladding were adopted. This temperature increase, of about 2C, results from “accepting” higher void fractions in the core subchannels, which is reasonably justifiable given the superior corrosion resistance of FeCrAl with respect to Zircaloy, and therefore the lower susceptibility of FeCrAl to CRUD formation. The increase in PHE temperature translates into an increase in plant thermodynamic efficiency, which plays a key role for the economics of the I²S-LWR concept.

6.8 Conclusions and future work

An assessment of the thermal hydraulic performance of the I²S-LWR core has been performed, covering steady-state full power operation and a Complete Loss Of Flow Accident. It has been demonstrated that, in order to approach core outlet temperatures similar to those of existing PWRs, and therefore aiming at similar if not higher plant efficiencies, the temperature at the inlet of the I²S-LWR core must be relatively high, around 298°C, which is however within operating experience. In order to meet selected thermal hydraulic constraints, especially low void fraction in the hottest subchannel and MDNBR during CLOFA, these relatively hot coolant conditions must be combined with a relatively high coolant velocity, about 33% higher than in a typical 4-loop PWR and 20% higher than in a high-velocity PWR. Although efforts will be made to reduce this velocity, a higher velocity with respect to existing plants may be acceptable given recent results obtained elsewhere, which seem to indicate FeCrAl to be more fretting resistant than Zircaloy ([3]).

The assessment also revealed that, for an U_3Si_2-fueled I^2S-LWR core, it is reasonable to expect a safety margin to fuel melting equal, larger or much larger than that for the same core however fueled with UO_2. The uncertainty in U_3Si_2 swelling behavior, and on the way Pellet-to-Clad Mechanical Interaction takes place for this fuel, has not allowed so far to focus on a single fuel rod design, which is why it is not possible to precisely indicate the increase in safety margin to melting to be expected with U_3Si_2. However, based on an analysis performed on some "bounding" U_3Si_2 pellet designs, it is reasonable to expect an increase in safety margin, with respect to UO_2, up to 2.3 times.

When additionally the credit is take for the improved oxidation resistance of FeCrAl clad, void fraction constrain may be relaxed to 0.3 (and perhaps even a higher value). This allows increasing the core outlet temperature, i.e. the PHE inlet temperature, by about 2°C, This increase in PHE temperature translates into an increase in plant thermodynamic efficiency, which plays a key role for the economics of the I^2S-LWR concept.

6.9 References

[1] P. Ferroni, B. Petrovic, G. Sjoden, M. Chin, Preliminary thermal-hydraulic feasibility evaluation of the Integral Inherently Safe LWR (I^2S-LWR) high power density core. Proceedings of ICAPP 2014. Charlotte, April 6-9, 2014.

[2] AP1000 Design Control Document, Revision 19, Chapter 4. Available at NRC electronic library, ADAMS accession number ML11171A446.

[3] T.S. Byun, Friction-wear Behavior of Advanced Cladding Materials. FCRD AFC Integration Meeting, August 28, 2014.

[4] Watts Bar Updated Safety Analysis Report, Amendment 108, Chapter 4. Available at NRC electronic library, ADAMS accession number ML12083A170.

[5] NUREG-1920, Vol.1. *Safety Evaluation Report Related to the License Renewal of Vogtle Electric Generating Plant, Units 1 and 2*. April 2009. Available at NRC electronic library, ADAMS accession number ML091320261.

[6] Westinghouse Electric Company, Columbia Fuel Fabrication Plant – *Application for Certificate of Compliance for the Traveller PWR Fuel Shipping Package*. Rev. 1, Chapter 6, November 2004. Available at NRC electronic library, ADAMS accession number ML043410121.

[7] B. Petrovic, F. Franceschini, P. Ferroni, *Fuel Cycle Cost Trade-off Studies for I^2S-LWR (Integral Inherently Safe LWR) Fuel Design Selection*. LWR Fuel Performance (TopFuel) conference. Charlotte, NC, USA. September 15-19, 2013.

[8] NP-2511-CCM. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 2: User's Manual. Revision 2. Prepared by Battelle, Pacific Northwest Laboratories for the Electric Power Research Institute. July 1985.

[9] AP1000 Design Control Document, Revision 19, Chapter 15. Available at NRC electronic library, ADAMS accession number ML11171A370.

[10] Watts Bar Updated Safety Analysis Report, Amendment 108, Chapter 15. Available at NRC electronic library, ADAMS accession number ML12244A088.

[11] S.L. DAVIDSON, W.R. KRAMER (Ed.), Reference Core Report VANTAGE 5 Fuel Assembly. WCAP-10445-NP-A (Non-Proprietary), September 1985.

[12] Advanced Light Water Reactor Utility Requirements Document, Rev. 8. Electric Power Research Institute (EPRI). Section 3.1.1.1 in Vol. 1. 1999.

[13] A.T. Nelson et al., Overview of Properties and Performance of U-Si compounds for LWR applications. Presentation given at the 2014 Nuclear Fuels and Structural Materials meeting (NSFM), Reno, Nevada, USA, June 19, 2014.

Part II – Topical Reports – Executive Summaries

7. Materials (Report I2S-FT-16-02)

Prepared by: Paolo Ferroni (Westinghouse Electric Company LLC); Chaitanya Deo, Matias Marquez, KkochNim Oh, Preet M. Singh, Thomas Winter, Bojan Petrovic (Georgia Institute of Technology); Abderrafi Ougouag (Idaho National Laboratory); Indrajit Charit (University of Idaho); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 180 pp.

This topical report Materials" summarizes results of the campaign on materials behavior, mainly experimental but also with analytical/modeling aspects, performed in support of the development of the I²S-LWR concept. In order to enhance its safety and economic performance, the I²S-LWR adopts enhanced-performance fuel and cladding materials, relative to the conventional UO_2-Zircaloy combination. For the fuel it adopts high-density, high-thermal conductivity uranium silicide (U_3Si_2). For the cladding, the I_2S-LWR is envisioned to use corrosion-resistant FeCrAl-type steel in the near-term, followed by transition to SiC/SiC composite in the longer term.

Specifically, the activities discussed in this report are the following:

- Section 1: Cladding corrosion testing
- Section 2: Cladding mechanical property testing
- Section 3: Cladding fretting wear and nanoindentation testing
- Section 4: Cladding-fuel compatibility testing
- Section 5: Development of swelling models for U_3Si_2
- Section 6: Materials properties database (Appendix)

<u>Cladding corrosion testing</u>

To address post Fukushima concerns, I²S-LWR implements Accident Tolerant Fuels, i.e. fuel/cladding materials capable to survive loss of cooling for a longer period of time compared to Zr-based alloys, without self-catalytic reactions, with reduced oxidation rates, and no or minimum hydrogen production. Experimental campaign was focused on confirming the superior oxidation resistance of advanced FeCrAl-type ferritic stainless steels representative of those envisioned for use in the I²S-LWR concept, e.g. APM and APMT. In addition to high temperature oxidation, the experimental campaign also evaluated the oxidation behavior under conditions representative of normal operation, i.e. aqueous environment and typical LWR operating temperatures.

High temperature oxidation behavior under 79% N_2+21% O_2 (dry air) condition, was examined using thermogravimetric analysis (TGA) and compared to that of ZIRLO® alloy[6]. ZIRLO® oxidation rate significantly increased upon an increase in the test temperature from 400 ºC to 800 ºC, and the sample failed after only 10 h at 800 ºC due to the severe oxidation. Oxidation rates of APM and APMT increased with temperature as well, but in contrast, at 1000 ºC and 1100 ºC were still significantly lower than

[6] ZIRLO is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world.

ZIRLO® at 600 ºC and 800 ºC. APMT showed lower rate constant over the whole temperature range between 600 ºC and 1100 ºC and had a lower activation energy than APM.

For high temperature oxidation behavior of APM and APMT under 100% steam condition at 1000 ºC up to 120 hours., oxidation rate was higher compared to the equivalent tests done under dry air condition. Moreover, the grain size and the thickness of oxide layer of APM and APMT under 100% steam condition were larger and thicker than those under dry air condition.

Additionally, immersion corrosion tests were performed under normal operating temperatures of several materials of potentials interest, with well controlled nominal water chemistry, and also under elevated levels of dissolved oxygen. Immersion tests under nominal chemistry were performed in double distilled ionized water (D-DI water) and in deaerated double DI-water (DD-DI water) at 320 °C for 30 days, These tests provided the following findings:

- Under all water conditions, APM and APMT showed the best corrosion resistance of all tested alloy, and was significantly better than the ZIRLO samples tested under similar conditions.
- Corrosion rates generally increased with an increase in the test temperature and presence of oxygen, especially for the ZIRLO.
- Presence of oxygen (with 7.7ppm dissolved oxygen) in doubly deionized water (D-DI) resulted in significantly higher corrosion rates for ZIRLO and low chromium alloys compared to the corrosion rates found in deaerated doubly deionized (DD-DI) water (with < 1 ppb dissolved oxygen). APM and APMT showed the lowest corrosion rates.
- Corrosion rates for APM and APMT alloys were virtually negligible at test temperatures of up to 350°C.

In summary, APM and APMT are very resistant to corrosion under reactor water operating conditions, more so than the ZIRLO; they are also much more forgiving to the water chemistry control upset, as the corrosion rates do not significantly change with minor changes in the dissolved oxygen concentrations.

<u>Cladding mechanical property testing</u>

The main objective of the was to study the mechanical properties of the aluminum-rich ferritic alloys (APMT and Fecralloy) over a temperature range, 25 to 500C, not completely covered by literature data, and compare such properties to those of Zircaloy. Tensile testing allowed determination of important mechanical properties such as yield strength, ultimate tensile strength, and percentage elongation to fracture. Vickers microhardness testing (Vickers hardness) and metallographic studies (microstructural characteristics) were also conducted.

Microhardness testing provided the following results:

- APMT rod exhibited a higher hardness than FeCralloy rod. Zircaloy-4 hardness was the lowest among all materials tested
- APMT tube showed a greater hardness value then APM tube.
- Mechanical anisotropy as determined from the hardness testing indicate that Zircaloy-4 is more anisotropic than Al-rich ferritic steels.

Tensile tests provided the following results:

APMT™ steel received in the rod form shows quite high yield strength of >500 MPa at all test temperatures and strain rates. The ultimate tensile strength was found to be >650 MPa at all temperatures. The trend in ductility was not uniform but remained above 20% (elongation to fracture) under almost all conditions of testing. Serrations are observed in the stress-strain curves at the intermediate temperatures in certain strain rate ranges due to dynamic strain aging. The solid solution strengthening imparted by Mo and subgrain strengthening account for the higher yield strength of Kanthal APMT™ steel than that of Goodfellow FeCralloy™.

Cladding fretting wear and nanoindentation testing

Relative motion between the fuel rods and fuel assembly spacer grids can lead to excessive cladding wear and, potentially, to fuel rod failure. Grid-to-rod-fretting (GTRF) has been a significant cause of fuel failures within the U.S. pressurized water reactor (PWR) fleet, accounting for more than 70% of all PWR leaking fuel assemblies. An experimental campaign was therefore conducted to specifically address this phenomenon, especially in consideration of the peculiarities of I²S-LWR, namely novel cladding materials (FeCrAl-type and SiC/SiC) and a relatively high coolant velocity. While these novel cladding materials are anticipated to perform better than Zircaloy from the corrosion standpoint during loss-of-coolant events, it is important to ensure their reliable performance during normal operation by assessing, among other things, their GTRF performance.

Tests were performed to examine behavior of the I²S-LWR cladding candidates under simulated fretting conditions of a pressurized water reactor (PWR). Numerous sample holders where developed to allow for simulated PWR wear conditions. APMT (representative of FeCrAl-type steels) and SiC/SiC claddings were investigated. A combination of SEM analysis, 3D Confocal microscopy, and wear & work rate calculations were performed on the samples to determine their performance and wear under fretting.

Cladding-fuel compatibility testing

The introduction of novel fuel and cladding materials requires their compatibility, in the event of pellet-cladding interaction (PCI), to be assessed. With this purpose, diffusion couples of various fuel and cladding materials were manufactured, tested at temperature up to 1000C, and subsequently examined using, mainly, Scanning Electron Microscope (SEM). Thermodynamic calculations were also performed on the stability of the U3Si2-SiC system. Three main conclusions were drawn:

1) Compatibility was generally good for the couples tested. No large-scale reactions were observed following exposure for temperatures up to 1000 °C and time intervals below 100 hours. During actual reactor operation, contact between the cladding and fuel materials is expected to last much longer than 100 hrs (e.g. months to years), and some species interdiffusion is anticipated. However, based on the experimental results collected, rod failure due to the formation of low melting point phases is not anticipated.

2) Formation of an U-Al-O phase appears to be a possibility for U_3Si_2 and Fe-Cr-Al alloys. Furthermore, qualitative investigation suggests that U3Si2 may act as oxygen getter more effectively than candidate Fe-Cr-Al materials. The effect of this on reactor operation should be investigated.

3) When in contact, U_3Si_2 and SiC are expected to experience phase instability, with formation of uranium carbide and of different uranium silicide phases (i.e. USi and U_3Si_5). However, these thermodynamic calculations do not provide information on the reaction kinetics, which may be very slow. Follow-on work, beyond the I_2S-LWR program workscope, will investigate the kinetics of these reactions to evaluate the potential design impact.

<u>Development of swelling models for U_3Si_2</u>

One technical challenge in the I^2S-LWR design is related to the uncertainty in the silicide fuel swelling. It is generally believed that it may be higher than that for oxide fuel, but it was not known by how much. This is a critical performance parameter that may significantly impact fuel design, or even the feasibility of using silicide fuel. High swelling would require use of annular fuel and a larger fuel-to-clad gap, which would eventually negate the benefits of silicide fuel. Therefore, during the course of the projects, silicide swelling was investigated by two groups within the project team. Both groups examined fission gas bubble formation and U_3Si_2 fuel swelling in bulk material.

Most of the experimentally available silicide swelling data were obtained at relatively low temperatures in research reactors. Under these conditions, silicide becomes amorphous quickly and remains amorphous up to a fairly high fluence. The first group therefore modeled swelling for amorphous silicide fuel which allows comparison to the experimental data; good agreement was observed.

The second group assumed instead crystalline fuel form, representative of I^2S-LWR operating conditions, with grain subdivisions impacting fuel swelling The model does not account for any fission gas release, and assumes that recrystallization, if it occurs, will occur over the whole pellet volume, which produces conservative swelling estimates. Since there are no experimental data obtained under similar conditions, by necessity it used educated guesses for several parameters. Actual values may be different, impacting the swelling estimate. Ongoing silicide irradiation program at ATR may help to improve the values of these parameters. Keeping all these caveats in mind, the results indicate recrystallization around 50 GWd/tU, with acceptable swelling below that burnup, and unacceptable above. This is just about the average predicted discharge burnup for I^2S-LWR. However, fission gas release should further increase this critical burnup value, and swelling of silicide in I^2S-LWR is therefore expected to be acceptable. Experimental work at ATR is expected to confirm or disprove this assessment.

<u>Materials properties database</u>

To be able to consistently and accurately evaluate various fuel/clad options, an extensive literature search was performed, and a database with material properties, for all fuel/clad materials of interest was prepared based on a critical literature review to identify the most credible values. This review also contributed to identifying the gaps, and consequently helped define a limited but targeted set of experiments described in this topical that were performed within this project.

Further details are provided in the topical report “Materials”.

8. Core Design and Performance (Report I2S-FT-16-03)

Prepared by: P. Ferroni, F. Franceschini, D. Salazar (Westinghouse Electric Co., Cranberry, PA); B. Petrovic, F. Rahnema, Ce Yi, D. Zhang, K. Ramey (Georgia Institute of Technology); D. Kotlyar, G. Parks (University of Cambridge, Cambridge, UK); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 148 pp.

The topical report "Core Design and Performance" first discusses the viability of a high power density (HPD) core, then presents core designs for a range of fuel/clad systems (oxide and silicide fuel; ZIROLO, FeCrAl and SiC cladding) and refueling options (12-month and 18-month cycle), together with a fuel cycle cost analysis. Impact of the radial neutron reflector and advanced Monte Carlo and response-matrix methodologies needed for accurate core performance evaluation are documented as well.

Specifically, report section include:

- High Power Density Core: design and thermal-hydraulic assessment
- Flow Induced Vibration Analysis
- Equilibrium Cycle Core Analysis
- First Core Design for 18-month cycle
- Stylized I^2S-LWR Benchmark for Cross-Verification of Codes and Methods
- Serpent Based Methodology for Full Core Depletion Simulations

One of the key areas of the project was to address the viability of the I^2S-LWR high power density (HPD) core, to develop representative reloading strategies and associated core designs, and perform fuel cycle cost (FCC) analysis. Requirement on the HPD core was to provide the same or enhanced safety and performance as the current standard PWR cores. Reloading strategies should provide 18-month and 12-month refueling option, with viable equilibrium and first core designs. For the 18-month refueling, an advanced first core (emulating the equilibrium cycle core) is also sought. In all cases, use of the several fuel/cladding systems should be considered, from the traditional potentially start-up UO_2/Zirc fuel, to enhanced U_3Si_2/FeCrAl in mid-term, and ultimately U_3Si_2/SiC fuel expected to provide the best safety and economics performance.

<u>High power density core</u>

First task was to assess the viability of the I^2S-LWR high power density (HPD) core, i.e., to design a core capable to generate and safely remove a volumetric power density about 20% higher than conventional PWRs. A reduction in fuel rod linear power, resulting from an increase in the number of fuel rods per unit of core cross sectional area relative to typical PWRs, has been a key design choice for achieving this objective. The use of accident tolerant materials contributes to further enhancing reactor performance, with U_3Si_2 fuel providing higher fissile content and enhanced thermal properties relative to UO_2, thus benefitting FCC and reducing fuel operating temperature, and FeCrAl-type cladding improving corrosion performance and thus significantly reducing hydrogen generation. Design efforts were not limited to the U_3Si_2-FeCrAl material combination, which represents the reference design, but were extended to other material combinations in order to properly and comprehensively address all the possible phases envisioned for the I²S-LWR deployment. Specifically, the analysis considered a "first cores assembly design" featuring conventional UO_2 fuel and Zircaloy cladding, and an "enhanced performance design" using U_3Si_2 fuel and SiC/SiC cladding. The latter material combination, in particular, is anticipated to achieve best performance in terms of safety but especially economics, in consideration of its capability to fully exploit the high fissile content of U_3Si_2 fuel without introducing the neutron economy penalty that instead characterizes the FeCrAl cladding. The choice of the same fuel rod pitch and outer diameter, and of the same assembly envelope (axially and radially), allows the I²S-LWR to transition from one core configuration to the other while limiting the number of changes to reactor internals. Together with the companion neutronics analysis, a preliminary thermal-hydraulic assessment demonstrated that the U_3Si_2-FeCrAl and U_3Si_2-SiC/SiC designs have great potential for allowing the I²S-LWR to simultaneously meet the power output and safety requirements established for this novel reactor concept. The key parameter used in analyses was the "overpower-to-melt". In the complete loss of flow accident (CLOFA), the main assumptions were on the coolant outlet temperature and velocity, and the pump inertia, i.e., coast-down characteristics. With adequate choices, the overpower-to-melt margin in I²S-LWR ranges from slightly better to significantly better compared to a representative PWR with 17x17 standard fuel. Thus, a HPD core can be designed to provide satisfactory thermal and safety performance.

<u>Flow induced vibrations</u>

In addition to thermal performance, the HPD core – due to its increased coolant velocity – may also challenge the mechanical integrity of fuel. Therefore, potential susceptibility and limits with regard to flow induced vibrations (FIV) were examined. Comparing traditional Zircalloy based oxide fuel to APMT (FeCrAl) clad silicide fuel using an assumed requirement for high axial flows (6 m/s), it was found that overall, the silicide fuel offered more conservative resistance to FIV when considering fluid-elastic instability, vortex shedding, and axial flow instability thresholds. The grid-to-rod fretting wear analysis shows that the I²S-LWR U_3Si_2/FeCrAl 19x19 fuel design has similar characteristics with respect to FIV of the reference PWR UO_2/Zirc4 17x17 fuel design. While this assessment is based on average parameters, and not a substitute for a detailed CFD simulation and dynamic work rate model for grid-to-rod fretting wear evaluations, the approach used in this study is expected to provide correct relative results, suggesting adequate performance of the I²S-LWR fuel, in spite of the more challenging environment.

<u>Equilibrium cycle core design for 12-month refueling</u>

A 12-month equilibrium cycle core design has been devised for the I²S-LWR which implements an efficient fuel management scheme while satisfying top-level safety limits, including peaking factors and shut-down margin (SDM). The fuel design is a 19×19 square lattice U_3Si_2 fuel pellet, initially considered with an inner void, later as a solid pellet, in advanced FeCrAl steel cladding. The fuel active length is 144-in with top and bottom axial blankets. IFBA is used as the fuel burnable absorber. The fuel management is a 3-batch split feed with ^{235}U concentrations of 4.45 and 4.65 w/o, and 2.6 w/o ^{235}U blankets, for an assembly-average discharge burnup of ~40 GWd/tU and of ~50 GWd/tU for the peak pin. A core design implementing UO_2 fuel with FeCrAl steel cladding has also been developed, as a potentially shorter-term alternative to U_3Si_2. From the core design standpoint, the main differences compared to the U_3Si_2 core is the higher enrichment, 4.80 w/o ^{235}U in UO_2 vs. an average of 4.53 w/o for the full-enrichment axial region, and 3.2 vs. 2.6 w/o ^{235}U for the axial blankets. This higher enrichment is required to compensate for the lower heavy metal content of UO_2 vs. U_3Si_2. Some slight differences in the physics behavior are noted and ascribed to the slightly different neutron spectrum for the two core designs or to the higher operating temperature of UO_2 fuel (more negative Doppler power coefficient (DPC) in UO_2, higher SDM in U_3Si_2). None of these reactor physics differences constitutes a decisive advantage or disadvantage in determining the feasibility of either fuel option. In summary, U_3Si_2 and UO_2 appear both feasible options for the I²S-LWR 12-month cycle.

Equilibrium cycle core design for 18-month refueling

An 18-month equilibrium cycle core design has been devised as well, satisfying top-level safety limits, including peaking factors and SDM. The baseline fuel design is the same as for the 12-month refueling, with U_3Si_2 fuel pellet in advanced FeCrAl steel cladding, and IFBA. The fuel management is a 2.3-batch with ^{235}U enrichment of 4.8 w/o, and 3.0 w/o ^{235}U in blankets, for an assembly-average discharge burnup of 41 GWd/tU. A core design implementing UO_2 fuel with ZIRLO cladding has also been developed for accelerated deployment of the I²S-LWR concept. From the core design standpoint, the main difference compared to the U_3Si_2 core is the higher enrichment, 4.95 w/o ^{235}U in UO_2 for the mid-region and 3.2 w/o ^{235}U in the blanket region which compensates the lower U content of UO_2 vs. U_3Si_2 and reflects in the higher discharge BU, 51 GWd/tU. Some differences in the physics behavior are noted and ascribed to the harder neutron spectrum in U_3Si_2 or to its lower operating temperature compared to UO_2 fuel (more favorable radial power peak in U_3Si_2, more negative DPC in UO_2, larger shut-down margin in U_3Si_2). None of these reactor physics differences constitute a decisive advantage or disadvantage in determining the feasibility of either fuel option. The more efficient U usage, mostly from lower parasitic captures in ZIRLO vs. FeCrAl, should be noted. This reflects in ~17% lower FCC of UO_2/ZIRLO vs. U_3Si_2/FeCrAl. Finally, a core design based on U_3Si_2 fuel in SiC cladding has been developed operating on a 2.5 batch fuel management scheme, with 4.78 average ^{235}U enrichment and 53 GWd/tU average discharge burnup. It results in intermediate reactor physics characteristic between the two other cores, and definite fuel cycle cost advantages: 4% lower fuel-related electricity cost than the UO_2/ZIRLO and 20% lower than U_3Si_2 with FeCrAl cladding. The lower fuel cost compared to UO_2 derives from the higher U content and better fuel management options that it allows. The fuel cost advantages compared to U_3Si_2 with FeCrAl cladding are mostly due to significantly lower parasitic captures in the SiC cladding with the more favorable H/U as a secondary contributing factor. In summary, U_3Si_2 with FeCrAl or SiC cladding as well as UO_2 with ZIRLO cladding appear all feasible options from a core design perspective.

Advanced first cycle core design

Traditionally, the first cycle core design is based on a 3-batch out-in loading scheme. Several cycles (4-6) are subsequently needed to transition to the equilibrium cycle which is typically based on in-out loading scheme. This transition introduces some FCC penalty. Here, for the 18-month refueling, an advanced first core is sought, that would emulate the equilibrium cycle core, and enable much faster (in 2-3 cycles) and economically advantageous transition to the equilibrium cycle. This was achieved by matching the reactivities of the fresh fuel to those of fuel in the target equilibrium cycle. An accurate match required a large number (30) of different enrichments of the fresh fuel, which may be impractical. However, an approximate match with significantly reduced number of different enrichment zones (12) was developed that provided very similar performance. In either case the core performance (e.g., critical boron concentration and peak factors) was in the first cycle reasonably close to that of the target equilibrium cycle, and already in the second cycle was essentially the same as in the target equilibrium cycle.

Cross-verification of reactor physics codes and methodologies employed by the team members

Several groups within the project team performed core physics analyses. Therefore, a benchmark was developed to facilitate cross-verification of reactor physics codes and methodologies employed by various team members. Multigroup multi-parameter cross-section libraries were generated using HELIOS for each unique I^2S-LWR assembly regions. The I^2S-LWR core design was stylized to create a simplified version of the equilibrium cycle core as a benchmark. The fission source convergence in Monte Carlo (MC) calculations in the stylized I^2S-LWR benchmark problem was analyzed, by comparing the axial fission density distributions estimated by 50 independent MCNP5 calculations with their averaged result as well as with the COMET solution. The deviation of the axial fission density distribution predicted by independent MCNP runs from the average of the 50 runs is two orders of magnitude higher than the reported MC statistical uncertainty. Thus, a large number of independent runs would be necessary to obtain a converged solution. In contrast, it was shown that the COMET results agree very well (within one standard deviation) with those obtained by averaging the results from the 50 independent Monte Carlo cases. Therefore, COMET can provide detailed reference solutions (e.g., pin-powers) to whole-core (large eigenvalue) problems, which would otherwise be computationally prohibitive using direct Monte Carlo simulations.

Monte Carlo full core simulations using Serpent

Analysis of the radial neutron reflector requires use of the neutron transport theory to obtain accurate results. For that purpose purposes, Serpent code was selected due to its focus on reactor physics problems and a full core and quarter core models developed. Analysis of the radial reflector effectiveness was performed for a range of reflector compositions (volume fraction of the cooling channels), and soluble boron concentrations. The best neutron economy is achieved for a steel-only reflector. Inclusion of the cooling channels reduces the reflector effectiveness, but it is acceptable up to about 10 vol%. Moreover, the Serpent model was extended to depletion simulations, with a single-channel T-H feedback capability, for future benchmarking of coupled neutronics/T-H deterministic analyses.

Further details are provided in the topical report “Core Design and Performance”.

9. Thorium-based Plutonium Incineration (Report I2S-FT-16-04)

Prepared by: D. Kotlyar, G. Parks (University of Cambridge, Cambridge, UK); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 66 pp.

The topical report "Thorium-based Plutonium Incineration" documents results of evaluation of feasibility and performance of plutonium incineration in I^{2}S-LWR using thorium-bearing fuel. Plutonium incineration is of significant interest to the UK since it has the world largest stockpile of plutonium. This task was performed by the UK team member, University of Cambridge, at no cost to the project. While the use of thorium is currently not considered by the U.S. DOE, it is useful to assess potential of I^{2}S-LWR for that application for future and/or worldwide application. At the start of the project it was anticipated that some features of the I^{2}S-LWR design (such as the advanced steel cladding) would support enhanced plutonium incineration; the purpose of this task was to investigate and quantify it.

Current experience of plutonium recycling is mostly limited to mixed oxide U-Pu (MOX) fuel. This approach is not particularly efficient since Pu destruction is accompanied by simultaneous generation of new Pu from U^{238}. Therefore, use of Th-Pu mixed oxide (TOX) fuel to increase Pu incineration efficiency was proposed by different authors, demonstrating enhanced Pu consumption by transitioning to the TOX cycle. The majority of these studies investigated the utilization of Pu fuel in standard or modified LWR cores assuming irradiation periods of about 50 MWd/kgHM. This limit on the discharge burnup is imposed primarily by the performance of Zircalloy cladding as its mechanical properties degrade with burnup.

In the I^{2}S-LWR design, a ferritic alloy was envisioned as the cladding material. It is expected that such cladding materials can withstand longer irradiation periods with much lower degradation of their mechanical properties than standard Zr alloys. Transitioning from Zr to ferritic alloys offers the opportunity to improve the economic performance of the plant by enabling longer irradiation periods, potentially reduced reprocessing, and ultimately higher incineration rate.

However, since longer fuel cycles would require uranium enrichment to exceed commercially available 5 $^w/_o$ of U^{235}. Use of TOX with in situ breed-and-burn (U^{233} continuously produced from neutron captures in Th^{232}) can provide the reactivity to extend cycle length and offer an alternative solution.

This report is structured as follows. First, the investigation of the ThO_2-UO_2 fuel cycle to improve resource utilization is presented. In order to reach improved performance the enrichment must exceed 20% and therefore this cycle was eliminated from considerations. Then, the feasibility of the Pu-Th (TOX) fuel cycle is demonstrated and its performance with respect to plutonium incineration is compared against the U-Pu (MOX) fuel cycle. These fuel cycles are also compared with respect to their margin to fuel melting by performing hot channel analysis. The research then focuses on loading pattern optimization with Simulated Annealing (SA) studies that seek to enhance the performance of the TOX cycle even further by increasing the number of fuel batches and for each adopting an optimized loading pattern (LP). Lastly, sensitivity studies regarding the various parameters that contribute to the efficient incineration of Pu and TRU are presented. The chosen parameters included different clad types, reactor grade Pu vectors, PuO_2 volumetric fractions and moderator-to-fuel volume ratios. SA method was applied to 24 different core designs to identify the most favorable LP for each design with respect to cycle length. Post-irradiation characteristics, such as radiotoxicities and decay heat for the various designs are also presented.

The results presented here indicate that more than 75% (51%) of Pu (TRU) could be incinerated while preserving the required safety limits. The results also indicate that achieving this high incineration allows to considerably reduce the decay heat power and cumulative energy after the ultimate disposal. This implies that the size of the repository could be reduced or alternatively more waste could be stored in a given space.

Further details are provided in the topical report "Thorium-based Plutonium Incineration".

10. Reactor Vessel Layout and Internals (Report I2S-FT-16-05)

Prepared by: Bojan Petrovic (Georgia Institute of Technology); *With Contributions:* Matthew Marchese, Tim Flaspoehler, Alex Huning, Dan Kromer (Georgia Institute of Technology); Matthew Memmott (Brigham Young University); Guy Boy (Florida Institute of Technology); Subject experts (Westinghouse Electric Company, LLC); Annalisa Manera (University of Michigan); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 164 pp.

This topical report documents reactor I²S-LWR reactor vessel layout, including internals. The vessel houses the reactor core with the bottom support plate and top plate, core barrel, radial neutron reflector, internals, internal CRDMs (iCRDM), primary heat exchangers (PHE), passive decay heat removal system (P-DHRS), pressurizer integrated in RPV head, and reactor coolant pumps, physically partly outside the vessel, but functionally part of the primary boundary.

Adequate space, access and clearance need to be provided for installation, maintenance and repair of all components, which may present challenges. Therefore, it was decided to develop a detailed preliminary layout early in the project, to enable assessment of these issues. Work to develop the vessel layout started in January 2013, one month before the formal start of the project, through a senior design project at Georgia Tech. The student design team did an impressive job developing a detailed layout during the Spring 2013 semester, which helped to jumpstart the project. This initial design underwent a number of limited modifications and adjustments, through several design iterations over the course of the project, but the overall features of the layout changed little. Analyses were performed at different times using the then-current RPV layout. Limited resources did not allow re-doing all analyses from scratch every time any modification to RPV was made. If it was estimated that the RPV modification would not significantly change the quantitative results, and would not impact the conclusions of previous analyses, only new analyses were performed. As a result, there is not a single complete set of results for the final RPV layout. Instead, this report contains documents capturing RPV layout design iterations and presenting corresponding results.

The vessel diameter is 4.90m O.D. and 5.42m OD, and its height is 22.78m. This size satisfies the manufacturability requirement, since the diameter is about the same as that of the EPR vessel. The layout model includes all components and internals developed to a fairly high level of detail. This topical report also describes some of the related analyses:

- Reactor coolant pumps (RCP) sizing
- Vessel stress analysis due to RCP mounting
- Analysis of variation in pressurizer water level

Finally, a human centered design (HCD) approach was applied to RPV design. This was a very limited-scope effort to evaluate potential benefits of this approach. The results are very encouraging, and any future efforts will consider incorporating HCD, if practical.

Further details are provided in the topical report "Reactor Vessel Layout and Internals".

11. Power Conversion System (Report I2S-FT-16-06)

Primary Heat Exchangers Section Prepared by: Daniel Kromer, Alex Huning, Srinivas Garimella (Georgia Institute of Technology); *Flash Drums and Steam Generating System Section Prepared by:* Matthew Memmott (Brigham Young University); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 112 pp.

This topical report presents design and performance analysis and optimization of the I²S-LWR Power Conversion System (PCS). Since I²S-LWR employs a novel PCS concept, analysis and optimization was combined with experimental testing to assess and confirm viability and practicality of the concept.

Current loop-type PWRs use piping (primary loop) to transfer the coolant, that has been heated up in the core, to steam generators, located outside the reactor pressure vessel (RPV). Steam is generated on the secondary side of steam generators and transferred to turbine to generate power. In contrast, most PWR SMRs implement an integral primary circuit configuration, where all the primary loop components, including the steam generators, have been relocated into the RPV. Other than their location, steam generators still function the same way, as heat exchangers with reactor coolant, in liquid phase on the primary side, and steam generated on the secondary side. Helical-coil steam generators have been frequently proposed in various SMRs, but other steam generator designs have been considered as well.

A significant novelty was introduced in the Westinghouse SMR design, with steam generators "split" into a recirculating steam generator inside the vessel, and a steam drum outside the containment. This concept is further expanded and developed in I²S-LWR, where the steam generator is replaced by a steam generating system (SGS), composed of primary heat exchangers (PHE)within the RPV, and flash drums used to generate steam outside the containment.

In I²S-LWR, PHEs operate in single-phase liquid-liquid mode, and are of microchannel type. Both of these features contribute to PHE compactness, i.e., high volumetric power transfer capability. This is one of the main technological novelties that enables increasing the reactor power of I²S-LWR to 1 GWe, several times more than a typical SMR, while at the same time keeping the integral RPV size within the currently existing manufacturing limits. (The vessel is of similar size to that of large PWRs, e.g., EPR and APWR.) The steam generated in flash drum drives a multistage turbine system, completing the PCS. While this novel PCS system allows a more compact RPV, it is also more complex than the traditional use of steam generators and therefore requires careful optimization to achieve desired performance.

The report is organized as follows. Chapter 2 presents design of PHE based on microchannel heat exchangers (MCHX). An experimental scaled-down test facility was built to measure key performance parameters and verify predictions obtained by numerical simulations. Main challenge was meeting the performance requirements within the limited space defined by the RPV annular downcomer, which required careful optimization. Chapter 3 presents introduces the flash drum design, then devotes most of the efforts to discuss optimization of the power conversion system comprised of the primary heat exchangers, flash drums, and multi-stage turbine. Optimization of this complex system involves carefully balancing all components to achieve the desired performance level. Chapter 4 summarizes high level outcomes and conclusions. Main results include:

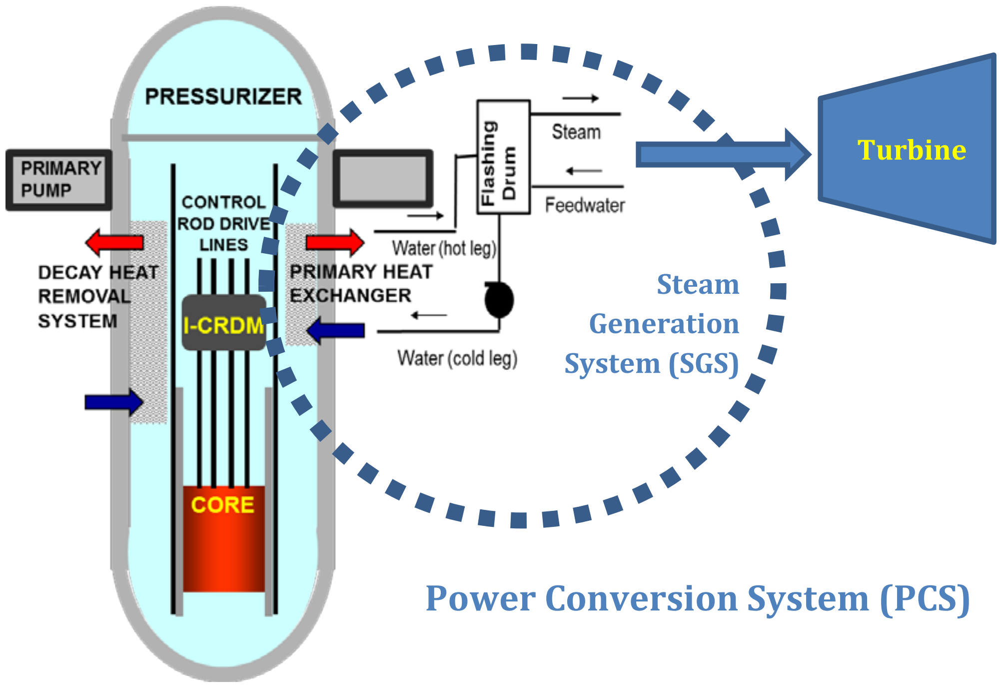


Figure 11-1: Power conversion system (PCS) of I²S-LWR

A microchannel heat exchanger (MCHX) design that can meet the requirements was selected. The MCHX consist of sheets with 445 of small diameter channels, D_h = 0.812 mm, chemically etched onto their surface. A total of 530 sheets, 1.13 mm thick, are stacked, alternating between primary and secondary, and diffusion bonded to create the heat exchanger unit. Individual heat exchangers, each with dimensions of 1 m × 0.85 m × 0.6 m, are stacked 11 units high and attached to headers designed to distribute the two fluids into the microchannel arrays. Eight such heat exchanger stacks (88 HX units in total) and associated headers are located around the core barrel in the downcomer.

A model of the MCHXs in the core was developed using the Engineering Equation Solver platform. Heat transfer and pressure drop correlations developed for larger hydraulic diameter channels were utilized and the literature was surveyed to estimate thermal resistance from fouling deposits in the channels. The ASME boiler and pressure vessel code for plate stayed pressure vessel was used to determine minimum wall and sheet thicknesses. An analysis of channel and header dimensions was performed to determine the optimum geometry. A parametric study of MCHX operating conditions was then performed to determine the maximum thermodynamic efficiency that could be achieved using the flash drum coupled Rankine cycle. This resulted in an efficiency of 34.6% producing approximately 986 MW_e. This has been further optimized in conjunction with flash drums.

A facility was constructed to test the performance of an MCHX representative of that used in the I²S-LWR design. Experimental results showed excellent agreement with both heat transfer and pressure drop prediction from the model. In addition, friction factor and Nusselt number correlations also showed good agreement in the turbulent regime in which the I²S-LWR MCHX will operate. These results confirmed the modeling capability and demonstrated viability of the proposed concept

The SGS design analysis considered three configurations. The first configuration (case 1) is the simplest, with only one high pressure (HP) turbine, a turbine reheat stream with a subsequent heat exchanger, and three low pressure (LP) parallel turbines. This cycle option has the lowest thermodynamic efficiency, but it conversely has the lowest capital cost requirements. The second configuration (case 2a) is similar to the first, except that an intermediate pressure (IP) turbine is included between the reheater exchanger and the 3 low pressure turbines. The third configuration (case 2b) represents the case with the potential for the highest thermodynamic efficiency with a similarly increased capital cost. This case is similar to case 2a, with the addition of a second reheater exchanger and commensurate reheat stream drawn from the intermediate turbine. One of the key parameters is the temperature of the fluid entering flash drums. This is the exit temperature on the secondary side from MCHX. Higher temperature is general expected to provide higher efficiency.

Initial parametric optimization identified sets of parameters that for case 2b resulted in efficiency of 35.7% and 35.9%, for flash drum inlet temperature of 319 and 321C, respectively. As expected, both cases2a and 2b provide better efficiency than case 1, and case 2b offers higher efficiency than case 2a. Therefore, case 1 was eliminated from further considerations. After correcting for the temperature-dependent heat transfer coefficient, the maximum efficiency was reduced, for case 2a and 2b to 34.58% and 34.70%, respectively. Moreover, rather than monotonically increasing with temperature, efficiency now peaks for flash drum inlet temperature around 319C. there is now an optimum range of temperatures.

Final multi-objective optimization was performed. It improved efficiency, to 34.974% for case 2b, essentially reaching the target 35%. Moreover, for the range of flash drum inlet temperatures between 317C and 319C, it is possible to achieve efficiency above 34.9%. Since the equipment costs seems fairly constant in that temperature range, the temperature that allows the highest efficiency, 318.3C, is selected as the reference, together with the set of parameters that produce that efficiency.

In summary, power conversion system has been designed that essentially provides the desired efficiency of 35% (34.974%). It incorporates microchannel heat exchangers as the primary heat exchangers, flash drums to generate steam, and a multistage turbine configuration (with high, intermediate and low pressure turbines, with corresponding reheaters). Design of microchannel heat exchangers has been developed. Additionally, experimental test facility was built at Georgia Tech to measure key parameters and confirm satisfactory performance of microchannel heat exchangers. Multi-objective optimization of the flash drum and turbine system has been performed to maximize efficiency of this fairly complex system, and a set of reference/optimum parameters has been determined. The main identified challenge is the need to use large pumps returning water from flash drums to microchannel heat exchangers. Overall, the reference design of the I²S-LWR power conversion system has been established.

Further details are provided in the topical report “Power Conversion System”.

12. Safety and Transient Analyses (Report I2S-FT-16-07)

Report sections prepared by: Annalisa Manera, Thomas Downar, Andy Ward, Mingjun Wang (University of Michigan); *With contributions by:* Matthew Memmott (Brigham Young University); Giovanni Maronati (Georgia Institute of Technology); *Report prepared by:* Bojan Petrovic (Georgia Institute of Technology); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 166 pp.

This topical report introduces the I²S-LWR safety philosophy, describes its safety systems, and presents results of safety and transient analyses.

As the first line of defense I²S-LWR employs safety-by-design to outright eliminate some accident initiators and corresponding accident sequences. This is followed by the defense-in-depth, which imposes multiple barriers to accident progression and aims to eliminate or reduce consequences of the remaining accidents. All safety systems are passive, which enables their functioning even under a station blackout (SBO), which is probably the most challenging event for current reactors. In particular, a passive decay heat removal system (P-DHRS) with a dry cooling tower using the ambient air as an unlimited ultimate heat sink, reliably ensures long-term (potentially indefinite) decay heat removal even under SBO, and with a single-failure assumption (3-out-of-4 trains operating). Similar P-DHRS is provided for the spent fuel pool cooling under SBO as well. Compact layout of the nuclear island makes its placement on seismic isolators technologically and economically viable. Use of accident tolerant fuel (ATF) provides additional coping time in severe accidents, either eliminating or postponing a large and early release (LERF).

Inherent safety features of the integral reactor vessel work synergistically with a compact containment that incorporates diverse passive safety systems (pressure suppression system, passive containment cooling system, passive reactor cavity cooling system, automatic depressurization system, accumulators and core makeup tanks). In severe accidents (beyond the design basis events) including SBLOCA, they provide alternate and redundant means of maintaining the integrity of the containment, or postponing LERF. This topical report focuses on deterministic analysis of design basis accidents (DBA), while a separate topical report presents results of PRA analyses.

Due to its significance, special attention was devoted to P-DHRS, considering sizing, design details, operational issues, etc. For all safety systems, functional requirements were defined and sizing performed, but in some cases the design specifics remained notional.

Accident scenarios typical of PWRs were considered and modified as needed to adequately represent I²S-LWR unique design features. Some accidents scenarios are eliminated by the I²S-LWR design itself (e.g., rod ejection and LBLOCA) and thus don't need to be analyzed. In some cases, several similar scenarios are represented with a single bounding case. Finally, a new accident scenario was introduced specific to I²S-LWR, secondary hot leg break (SHLB).

With this considerations, the following set of relevant and enveloping scenarios was defined and analyzed:

<u>Undercooling events:</u>

- Station Blackout (SBO)
- Loss of forced circulation (LOFC)

<u>Loss-of-coolant-inventory events:</u>

- SBLOCA (PZR valve stuck open)
- Spurious ADS actuation (SPADS)

<u>Reactivity insertion accidents:</u>

- Inadvertent rod withdrawn

<u>Overcooling events:</u>

- Secondary hot leg break (SHLB)
- Inadvertent actuation of DHRS

Analyses were focused on U_3Si_2/FeCrAl fuel, however, since the standard UO_2/Zircaloy fuel is also considered, primarily to enable accelerated deployment of I^2S-LWR and avoid new fuel licensing cost and uncertainty, some analyses were performed for that fuel type as well. Most accident scenarios required coupled neutronic/thermal-hydraulic analyses. In most cases coupled PARCS/RELAP5 were used, but when appropriate it was supplemented with CFD analysis.

Satisfactory safety performance was confirmed for all analyzed accident scenarios. In particular, I^2S-LWR provides long-term coping with station blackout, i.e., Fukushima-like accidents. Overall, I^2S-LWR enhances safety performance with respect to Gen-III+ LWR designs, and in particular address post-Fukushima concerns, i.e., provides core decay heat removal under SBO, spent fuel decay heat removal under SBO, extended coping time, and improved response to seismic events. Moreover, I^2S-LWR incorporates accident tolerant fuel in a holistic manner, with its design from the start harmonized with ATF, to achieve enhanced safety, without penalizing operation or economics.

Further details are provided in the topical report “Safety and Transient Analyses”.

13. Probabilistic Risk Assessment (PRA) (Report I2S-FT-16-08)

Prepared by: John Lee, Kellen McCarroll (University of Michigan); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 188 pp.

This topical presents probabilistic risk assessment (PRA) analyses and their use in the consept development process.

The I^{2}S-LWR PRA team has developed a thorough, Level-1 PRA for the I^{2}S-LWR conceptual design for internal initiating events. Throughout PRA development, improvements from risk, practical, and economic standpoints have been incorporated into the conceptual model, and thus the topical report describes the details of the plant developed by the PRA along with assumptions utilized at this stage. PRA results indicate the strong benefits of eliminating major initiating events, maximizing passivity, and balancing redundancy, diversity, and simplicity, yielding a low combined core damage frequency (CDF) and containment bypass frequency (CBF) of 1.01×10^{-8} per reactor-year. Multiple risk-important systems and components are identified and recommendations are given for further research and development activities related to plant safety.

Development of a preliminary PRA in parallel with plant design is a key facet of I^{2}S-LWR's safety-by-design approach. Together, the safety team has focused on elimination of initiating events (IEs), maximization of passivity in safety systems, and balancing redundancy, diversity, and simplicity of safety systems. The preliminary PRA now shows the benefits of the first two tasks and has been instrumental in striking an appropriate balance in redundancy, diversity, and simplicity.

This investigation of the I^{2}S-LWR preliminary PRA covers 14 IEs, including all design-basis accidents (DBAs) and several limiting transients. Reliability data were drawn extensively from the NRC documents, with "multiple-Greek letter" (MGL) parameters for common cause failures representations obtained from the EPRI Utility Requirements Document (Vol. III). . All passive safety systems are modeled, in most cases down to individual component-specific failure mechanisms. Several systems, however, are modeled with a single compound event based on information from the AP1000 PRA, courtesy of Westinghouse Electric Company. So far, only Level-1 analysis, focusing on internal IEs from full power proceeding to core damage, has been performed, although some sequences clearly lead to containment failure or bypass.

In total, the preliminary PRA includes 1854 basic events (BEs), 573 fault trees (FTs) linked to 1037 failure sequences organized into 25 event trees (ETs) mapping 14 IEs into 5 plant damage states (PDSs). The result is over 25,000 minimal cut sets, or distinct failure modes, resulting in a failure frequency estimate of 7.46×10^{-11}/ry, most of which is either core damage or containment bypass events. Combining these results with EPRI's recommended rate for reactor pressure vessel (RPV) ruptures of 1.00×10^{-8}/ry provides an extremely low plant damage frequency (PDF, sum of core damage frequency and containment bypass frequency) of 1.01×10^{-8}/ry.

As encouraging as the PDF estimate is, the true value of the I²S-LWR preliminary PRA is its ability to assess designs from a risk perspective and even focus design efforts. For example, interpretation of PRA results often shows that large portions of risk are associated with a single system or component. With this information, the design team is able to choose to modify, replace, upgrade, or augment the design. Additionally, two or more designs may be quickly evaluated in a risk-informed manner by modeling each in the preliminary PRA. To promote risk-informed design in such a manner, the PRA team has developed a technique to present these PRA results a digestible form, called simplified multi-path event trees (SMPETs).

The performed work should provide an excellent baseline for future implementation of more details (as the design further develops) and extension to PRA level 2 analysis, if the project continues.

Further details are provided in the topical report "Probabilistic Risk Assessment (PRA)".

14. Instrumentation and Control (I&C) (Report I2S-FT-16-09)

Prepared by: Belle Upadhyaya, Matthew Lish (University of Tennessee); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 158 pp.

This research focuses on the instrumentation and control of the Integral Inherently Safe Light Water Reactor (designated as I^2S-LWR). This ~1 GWe integral pressurized water reactor (PWR) incorporates as many passive safety features as possible while maintaining competitive costs with current light water reactors. In support of this work, the University of Tennessee has been engaged in research to solve the instrumentation and control challenges posed by such a reactor design. This report is a contribution to this effort. The objectives of this research are to establish the feasibility and conceptual development of instrumentation strategies and control approaches for the I^2S-LWR, with consideration to the state of the art of the field.

The objectives of this work are accomplished by the completion of the following tasks:

- Assessment of instrumentation needs and technology gaps associated with the instrumentation of the I^2S-LWR for process monitoring and control purposes.
- Development of dynamic models of a large integral PWR core, micro-channel heat exchangers (MCHX) that are contained within the reactor pressure vessel, and steam flashing drums located external to the containment building.
- Development and demonstration of control strategies for reactor power regulation, steam flashing drum pressure regulation, and flashing drum water level regulation for steady state and load-following conditions.
- Simulation, detection, and diagnosis of process anomalies in the I^2S-LWR model.

This report has evaluated the process measurements that need to be taken to safely and efficiently operate the plant, proposed means of taking those measurements, proposed strategies for controlling the operation of the nuclear steam supply system side of the plant based upon those measurements, and proposed means of monitoring the plant for anomalous conditions using established techniques in the field of fault detection and diagnosis.

To enable these approaches, the report presents the low fidelity dynamic modeling of the component systems for generating energy from nuclear fission and transporting that energy to a steam turbine for electricity generation. This modeling is necessary to examine how the different systems interact with one another in transient states and as the power output level of the plant changes, allowing for the examination of different approaches to controlling the plant. The models have further served to simulate scenarios of equipment degradation, reducing component performance. The simulated data have been used to develop and demonstrate approaches for monitoring the condition of these components and detecting and diagnosing these components when they degrade. This kind of automated approach will be critical in future nuclear power plants in order to reduce operation and maintenance costs to compete with advancing technology in other power generation sectors. The deployment and demonstration of these techniques may also serve to build a body of evidence which may lead to a reduced regulatory burden on plants operating with such highly sophisticated systems for monitoring plant health and performance.

The research has addressed significant instrumentation challenges faced by the I²S-LWR that involve:

- rapid response time measurement of primary coolant flow rate,
- placement of temperature sensors to accurately measure primary coolant temperature with acceptable signal to noise ratios,
- measurement of neutron flux level in the source and intermediate ranges, without having to replace the measurement equipment unacceptably often due to degradation during full power operation.

The report has proposed that the best candidate for measuring flow rate is the use of ultrasonic, reflection mode transit time flow meters mounted outside of the process, on the exterior of the RPV. Work needs to be done to develop this technology for application to large, thick vessels such as are typical for nuclear pressure vessels. Another promising candidate is noise analysis of signals from fixed in-core neutron detectors and core-exit thermocouples. This analysis can produce rapid response time data about the linear flow rate of coolant in various locations in the core, developing a flow profile and a measure of overall core flow rate. Fully submersible resistance temperature detectors, and housings, need to be developed for application to this reactor in order to avoid pressure vessel penetrations in the lower portion of the vessel, where they are prohibited as a design basis of the reactor. Silicon carbide neutron detectors utilizing lithium-six converting layers are suitable compromise detectors for placement in the downcomer region of the I²S-LWR, outside of the core, for safety related source and intermediate range neutron population monitoring. If placed at an appropriate radius within the downcomer, they achieve both necessary sensitivity and acceptable lifetime.

Dynamic modeling of the reactor systems produced stable, low fidelity simulation models suitable for feasibility studies of control, monitoring, and diagnostics. The models produced operate with stability and produce numerical results for steady states that are in agreement with the design basis operating points of the I²S-LWR. These models have been used to develop and evaluate control strategies for the reactor core, and steam flashing drum. Reactor core reactivity control is based upon a moving set-point for the average primary coolant temperature. This is done with a PID controller. The set-point moves between the average primary coolant temperature at full power and the saturation temperature of the secondary coolant at the drum operating pressure. This is because the drum pressure remains constant throughout operation, by design. Because of this, the temperature of the recirculation coolant in the drum must be the zero power temperature of both reactor coolants. When not producing power, the coolant flowing through the secondary side of the each exchanger, and not exchanging energy with the primary coolant, is at the drum operation temperature. For no exchange of energy to take place, the primary coolant must also be at this temperature. Implementation of this approach was demonstrated successfully for load following operation. Control of the flashing drum is achieved with two controllers. The drum level is maintained by adjusting the feedwater flow rate using a PID controller. The drum pressure, and thereby the steam pressure delivered to the turbine, is controlled by adjusting the steam flow rate using a throttling valve. Together, these two controllers maintain the flashing drum at the optimal condition to maximize the overall efficiency of the power conversion system.

The monitoring and diagnostics work has used data generated by simulating various kinds of equipment degradation in the plant model to demonstrate the applicability of established techniques for fault detection and isolation to the automated condition monitoring of the I²S-LWR. Detection of sensor drift affecting a control system, sensor drift by one of four redundant sensors, coolant flow reduction, and heat exchanger fouling have all been demonstrated.

Overall, this research is innovative and significant in that it reports the first instrumentation and control study of nuclear steam supply by integral pressurized water reactor coupled to an isenthalpic expansion vessel for steam generation. Further, this research addresses the instrumentation and control challenges associated with integral reactors, as well as improvements to inherent safety possible in the instrumentation and control design of integral reactors. The results of analysis and simulation demonstrate the successful development of dynamic modeling, control strategies, and instrumentation for a large integral PWR.

Further details are provided in the topical report “Instrumentation and Control”.

15. Shielding Analyses (Report I2S-FT-16-10)

Prepared by: Timothy Flaspoehler, B. Petrovic (Georgia Institute of Technology); Mario Matijević (University of Zagreb, Croatia); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 96 pp.

This topical report summarizes shielding studies performed at the Georgia Institute of Technology to support the design of I²S-LWR (Integral Inherently-Safe Light Water Reactor). The primary concept of any nuclear reactor design should ensure safe operation and minimize the dose to both the public and personnel while operating and maintaining the plant. While most design tasks are concerned with optimizing plant systems to create a safe and economically viable design, shielding studies primarily are done in support of other tasks to ensure that radiation produced as a byproduct of generating power is safely contained and its impact on materials and structures is minimized. Specifically, the scope within this project was to:

- Evaluate lifetime pressure vessel fast neutron fluence, to assess whether there are any embrittlement concerns.
- Evaluate gamma heating in the radial neutron reflector and consider the resulting cooling requirements.
- Evaluate the feasibility of placing the ex-core neutron flux monitors within the reactor vessel.
- Evaluate activation of primary heat exchangers and its impact on maintenability.
- Evaluate dose rate distribution in accessible areas outside the containment.

To demonstrate that all requirements are met, large detailed shielding models were developed within the Scale6.1 code package and used with the MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) shielding sequence to study the radiation environment surrounding the active core region. MAVRIC implements the CADIS and forward CADIS methodology which is considered to be the state-of-the-art shielding methodology in neutron transport. Generally, shielding calculations are very detailed and require large amounts of computer memory and long run times to achieve converged results, i.e., results with small statistical uncertainty in case of Monte Carlo simulations. For this reason, the University of Zagreb in Croatia performed independent analysis using their own models in order to support and verify results at Georgia Tech. These results are compared at the end of this report to help give confidence in the validity of conclusions made here.

Reactor Pressure Vessel Fluence: The first study performed was to envelope RPV (reactor pressure vessel) fluence over the lifetime of the plant to ensure safety in case of PTS (pressurized thermal shock) events. The RPV contains the primary system. Through the lifetime of the plant the RPV experiences a high fast neutron flux which causes neutron embrittlement. Embrittlement leads to an increased risk to PTS events. While, the I²S-LWR has a larger downcomer region which provides extra shielding to the pressure vessel, the design proposes a 100-year lifetime vs. a typical 40-year initial design lifetime for current operating LWRs. Also, a different fuel form and higher power density contribute to a different radiation environment and leakage to the RPV. Evaluation of the neutron fluence to the RPV was performed using both a low-leakage and high-leakage source description to envelope possible ranges of core designs. It was found that at the end of the 100-year plant lifetime the RPV would experience a maximum fluence below the traditional limit of $2(10^{19})$ n/cm^2 even with the high leakage core assumption. Additionally, throughout the design the downcomer region was increased which lessens the fluence by roughly 2 orders of magnitude. Therefore, PTS events should not be a concern for the I^{2}S-LWR but a surveillance program may still be necessary.

Radial Reflector Heating: The second study calculated gamma heating within the radial neutron reflector directly outside of the core. The I^{2}S-LWR design utilizes a steel reflector in order to increase the neutron economy by reflecting fast neutrons back into the peripheral fuel assemblies. However, steel absorbs the high energy gamma rays produced from fission which generates large amounts of heat. In order to keep the reflector at an acceptable temperature, cooling channels would need to be designed within the reflector to remove the heat. Adding water to the reflector degrades its efficacy in reflecting fast neutrons since the added water would thermalize them and significantly reduce the probability of return back into the core. To support the design of optimized cooling channels detailed gamma heating distributions were simulated. It was found that 5.95MW of heat is generated within the reflector during a steady state operation at nominal power. While this is a large amount of heat, it is possible to effectively remove it without significantly degrading the efficiency of the reflector in terms of the neutron economy.

Feasibility of ex-core, in-vessel Neutron Flux Monitors: Due to the wider downcomer region of the I^{2}S-LWR, typical power-level monitors that would be placed outside of the RPV would not have a high enough neutron flux to generate an acceptable signal. SiC monitors have previously been demonstrated for use in high-flux and high-temperature environment. A study was performed to determine SiC monitor placement within the RPV for both thermal and fast neutron responses. The tradeoff of placing them in the downcomer region is that the monitors would degrade over time if the flux is too high but if the flux is too low the response would be inadequate. The study used a range of boron concentrations in the primary coolant as well as a range of volume fractions of cooling channels in the neutron reflector. It was found that there is a region in the downcomer where the SiC power-level monitors could achieve an adequate signal while at a low enough flux to not have to be replaced too frequently, and potentially even lasting over the 100-year lifetime of the reactor.

MCHX Activation: One of the main novelties of the I^{2}S-LWR is integrating the entire primary system within the RPV which requires the placement of primary heat exchangers within the downcomer region inside of the RPV. In order to have a reasonably sized pressure vessel the heat exchangers needed to be compact. Therefore, the design chose to use MCHXs (micro-channel heat exchangers) which have a proven track record in other fields but have not been used for nuclear reactor applications. It is assumed that these would need to undergo maintenance operations throughout the lifetime of the plant, and being close to the core within the RPV results in the steel component becoming activated through ^{59}Co impurities. A shielding study was performed to calculate neutron activation of the MCHX and generate fixed gamma source terms to calculate dose rates around an exposed, dry MCHX. A method was developed which coupled detailed shielding calculations in MAVRIC with the ORIGEN-S activation sequence to accurately track all isotopic changes from neutron activation. Using the method, it was determined that activation within the MCHXs would not contribute a large dose to maintenance personnel. Additionally, after the 100-year lifetime the MCHXs would only be slightly more activated than the free-release limit (0.3Bq/g) as defined by the IAEA. This implies that the steel in the MCHXs could be recycled after 5 years as non-radioactive scrap metal.

Dose Rate Distributions: Finally, dose rate distributions during normal operations were calculated throughout a large nuclear island model both inside and outside of the CV (containment vessel). These are very large computations that push the limits of computational methodology and memory requirements. By using judicious mesh refinement for neutron biasing, converged dose distributions were obtained in regions outside of the CV and surrounding concrete shield. The maximum dose rate (9.4(10^{-9}) rem/hour) was found to be lower than background radiation levels, confirming that adequate shielding was used in the design.

Further details are provided in the topical report “Shielding Analyses”.

16. Containment Layout (Report I2S-FT-16-11)

Prepared by: Eric Yehl, (UC Berkeley; Engineering Analysis Intern at Westinghouse); Alex Harkness, Jay Schmidt, Matt Smith, Richard Wright, Paolo Ferroni (Westinghouse Electric Company LLC); Timothy Flaspoehler, Bojan Petrovic (Georgia Institute of Technology); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 44 pp.

This topical report presents the I^{2}S-LWR containment vessel design and systems arrangement. A 3D computer model of the containment layout has been produced with input from Westinghouse design experts. The model includes all components and features identified and required by the I^{2}S-LWR team, with many additional suggested in consultation with Westinghouse experts.

The containment layout and design incorporates top level requirements. The I^{2}S-LWR safety concept, building upon the IRIS and WEC SMR safety philosophy, requires a relatively small containment, that can take relatively high pressure in transients. In SB-LOCA, this enables quick equalization of the pressures in the vessel and containment, and stops the loss of coolant inventory from the vessel. The compact size is also beneficial for economics since it results in a compact footprint of the nuclear island. At the same time, the containment must provide sufficient space to fit all components/systems and to provide adequate access for maintenance. To combine this two opposing objectives and examine if/how it is feasible to achieve, it was necessary to develop a preliminary, physical layout, with sufficient level of detail and specificity. The layout incorporates all safety systems, which are sized based on the functional requirement established by safety analyses.

The containment layout effort was initiated as a GT senior design project, with guidance from Westinghouse experts and input from the safety Technical Working Group, led by the University of Michigan. The main bulk of the work was performed in the next iteration at Westinghouse, in combination with a student internship, resulting in the internal report: Eric Yehl, Alex Harkness, Jay Schmidt, Matt Smith, Richard Wright, Paolo Ferroni, "I^{2}S-LWR Containment Vessel Design and Systems Arrangement." That report has been integrated into this topical and expanded to provide additional information on dimensions and placement of the components.

The Containment Vessel (CV) is a large, pressurized cylindrical steel structure (23 m in diameter; with hemi-spheroidal top and bottom caps). containing the Reactor Pressure Vessel (RPV). CV serves as the last barrier (after the fuel cladding and RPV) to contain and prevent release of radioactive particles from being emitted to the environment during emergency scenarios.

The CV contains safety systems and components necessary for passive cooling during accident scenarios by removing heat and pressure from the RPV if primary (normal operation) systems are offline. ACC (Accumulators) tanks and CMT's (Core Makeup Tanks) provide borated water to the primary loop during accidents, such as, LOCAs (loss of coolant) events. PSS (Pressure Suppression System) tanks are used to condense steam released into the CV atmosphere from the RPV during accidents which would otherwise pressurize the CV to unacceptable levels. The PCCS (Passive Containment Cooling System) consists of a

HXs in the upper CV that act to remove decay heat (through a cooling tower) and help condense steam blown out of the RPV during accidents. The PRCCS (Passive Reactor Cavity Cooling System) is a helical coiled HX inside the reactor cavity that cools the RPV during accidents through the same cooling tower used for the PCCS. The CV layout also contains cranes, access hutches, space for maintenance, some (but not all) other components, and structural concrete.

One caveat is in place. CV vessel in the initial layout was 64 m tall, and this now outdated variant is shown in a number of figures in this report. During the last project year it became apparent that a shorter CV is required to produce the free volume consistent with its safety function. Moreover, shorter CV is also advantageous for economics and security (physical protection). PCCS was redesigned to enable the change, and an updated layout was obtained, with a 52 m tall CV. Resources were not available to re-generate all figures, and it was not deemed necessary since the change impacts only the top portion of the CV. Thus, only selected figures were updated, and some figures in the report still show the older, taller CV.

The model employs advanced parameterization tools, which make the layout arrangement extremely adaptable to design changes. Users can easily update critical dimensions in a list of variables, and the model will self-adjust to these changes using rigorously defined dimensional, geometric, and inter-relational constraints. In this manner, the size and shape of many components in the model can be updated effortlessly as the need arises.

At the end of the project, through a Georgia Tech senior design project, the updated layout was implemented into the 3D CAD model, and a 3D model of CV was printed.

The layout and information obtained from the 3D model was essential for safety systems analysis and design, as well as for shielding studies and economics evaluation. While certainly not a finalized design, the model is fairly well developed and incorporates all main features, as well as many details, and will be further developed if the I²S-LWR project continues.

Further details are provided in the topical report "Containment Layout".

17. Nuclear Island and Plant Site Layout (Report I2S-FT-16-12)

RT-TR-16-22 Prepared by: Robert M. Ammerman, Jerzy Chrzanowski, Alex Harkness, Paolo Ferroni, (Westinghouse Electric Company LLC); *Appendix A prepared by:* Bojan Petrovic (Georgia Institute of Technology); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 40 pp.

This topical report incorporates, with limited changes and additions, Westinghouse report RT-TR-16-22, issued as Westinghouse Non-Proprietary Class 3.

A study was conducted by an expert team at Westinghouse Electric Company LLC (Westinghouse) to create a general layout of the Integral Inherently Safe LWR (I²S-LWR) plant. The general plant layout described in the document includes layouts for the Nuclear Island (NI), Turbine Building, Radwaste Building, Annex buildings, and some additional structures and components located around those buildings in the "yard". Portions of the plant layout that are expected to employ same designs as a traditional PWR (such as the switchyard, administrative building, etc.) were not addressed in this study.

A 3D CAD model was created to assist in the layout, and to facilitate collaboration with other design partners. Equipment and other aspects that could potentially impact the layout of each building were evaluated and included in the 3D model. Details related to the conceptual layout for Heating, Ventilation, and Air Conditioning (HVAC) ducts, electrical raceway, piping, and supports for any of the aforementioned categories were omitted from this study, and are not represented in the layout or figures.

It should be noted that more time and emphasis was placed on the layout of the NI, which has characteristics specific to the I²S-LWR concept and as such more information is provided in the report relative to other buildings.

The Nuclear Island (NI) is sized to encapsulate the Containment Vessel (CV) and contains the components of the Primary Systems. The following systems are classified as Primary Systems:

- Reactor Coolant System (RCS)
- Chemical & Volume Control System (CVS)
- Residual Heat Removal System (RNS)
- Spent Fuel Pool Cooling System (SFS)
- Passive Core Cooling Systems (PXS)
- Passive Containment Cooling System (PCS)
- Liquid Radwaste Systems (WLS)
- Gaseous Radwaste Systems (WGS)

Specific to the I²S-LWR, the NI contains four large Flashing Drums, required for steam generation as the Primary Heat Exchangers (PHE), integral to the Reactor Vessel, operate in single phase on both primary and secondary side. Moreover, because of the small fraction of secondary liquid, less than 10%, that is actually flashed to steam, the current version of the Steam Generation System (SGS) features a large number of relatively large pumps, needed to send the secondary liquid that did not flash to steam, back into the PHE. Flashing drums and associated pumps occupy a significant volume, greatly increasing the overall size of the NI. Future optimization may be able to reduce that volume, and subsequently the NI size.

An array of seismic isolation devices is located below the bottom floor with space provided for access and inspection. Excluding the space required for seismic isolation, the current NI structure design measures approximately 63 m by 69 m (footprint), and 62 m high. It should be noted that, conservatively, relatively generous space was included in this layout for maintenance and repair. It is estimated that a more detailed optimization could reduce the footprint by about 10% in each horizontal direction.

The NI features seven main floors, excluding the roof, and three intermediate floors; only the top two floors are above grade. Key elevations relative to the grade are:

- Top of concrete for seismic isolation devices: -41.1 m
- Level 1 floor (top of concrete): -36.9 m
- Level 9 floor (top of concrete, plant grade): 0 m
- Roof (top of concrete): 16.8 m

This very low profile (elevation of the NI roof 16.8 m above the grade, and the containment vessel highest point around 23 m above the grade) provides enhanced physical protection, i.e., security-by-design.

While certain components and features have not been developed or implemented, the layout is quite detailed considering the pre-conceptual stage of the overall design. This has allowed a better-substantiated economic assessment of the construction cost already in this phase, and will serve as an excellent baseline to a more detailed layout, should the project continue.

The electronic 3D CAD model developed at Westinghouse was subsequently used in Spring 2017 for a senior design project at Georgia Tech to print a physical 3D model of the NI and to construct a 3D model of the whole site, as shown in the Appendix to the topical report.

Further details are provided in the topical report “Plant Layout”.

18. Spent Fuel Pool: Passive Decay Heat Removal System (Report I2S-FT-16-13)

Prepared by: Alessandro Banzatti*, Pietro Avigni, Bojan Petrovic (Georgia Institute of Technology), (*Georgia Institute of Technology, Politecnico di Milano, Milano, Italy); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 46 pp.

This topical report presents design of the spent fuel pool passive decay heat removal system, and results of analysis of its performance.

The overall goal of this study is the development of a passive cooling system for the spent fuel pool (SFP) of the I²S-LWR. This work defines a RELAP5-3D thermal-hydraulic model of the system and demonstrates its ability to passively maintain the reactor in safe conditions for at least two weeks (potentially indefinitely) following a Station Black-Out. To achieve this objective, this design takes advantage of natural convection of water on the primary and intermediate side and ambient air on the external side as ultimate heat sink, along with passive makeup water tanks and well-designed fuel assembly racks and pool layout.

The reference dimensions for the pool and the racks are taken from the AP1000 Design Control Document and adjusted to the I²S-LWR case; a conservative approach has been adopted in the selection process of dimensions and geometry configurations. AP1000 9x9 rack type has been selected as a reference and the pool is sized to accommodate 605 fuel assemblies, corresponding to a 3-batch fuel cycle, 12 third-core batches, plus a full core offload. A 4x2 rack layout has been chosen, and a 20 cm gap is left between the racks and the pool walls. The design also includes a multipurpose area for service and assembly maintenance, as well as service canals and pits, also designed to provide makeup water to the pool in case the level drops below the safety margins.

A secondary or intermediate circuit is necessary since the spent fuel pool will be positioned underground and the cooling tower is taller and located at higher elevation; a shell-and-tube straight tube intermediate heat exchanger has been selected in order to minimize the pressure drop in the natural circulation water loop. Three independent trains, sharing the same cooling tower, have been implemented in a 2-out-of-3 disposition, in order to fulfill redundancy requirements on reliability of residual heat removal capability.

A RELAP5-3D model of the spent fuel pool has been developed. The water-to-air heat exchange system (AHX) is open loop, 30 m tall, that allows the air to flow downward to the heat exchangers and then upward through the cooling tower.

Optimization has been conducted to maximize the effectiveness of the system in accidental conditions, using an iterative two-step procedure that optimizes the IHX keeping the features of the AHX fixed and vice versa in an alternated manner until convergence of the parameters is reached. The optimization process assumed that the system is in steady conditions and the power delivered by the assemblies is of the order of 12 MW; the maximum temperature for which a given design can be considered safe is assumed to be 90°C. Only the heat exchangers have been considered in the optimization process and the remaining parts of the system have been assumed fixed; the heat exchangers are both shell-and-tube straight-tube type with 1mm pipe thickness. a series of 53 simulations with different parameters produced the recommended internal pipe diameter equal to 8 mm for the IHX with 46765 tubes, and 21 mm with 11269 tubes for the AHX, respectively.

Analysis of a transient under a Fukushima-like event was conducted using the optimized RELAP5-3D model and simulating a SBO accident shortly after a full core offload (the most challenging SFP configuration). The decay heat is represented by two components, the first representing batches that have been sitting out of the core for at least one year, up to a maximum of 11 years, and the second component representing the last discharged batch plus the two remaining batches that are still being burned in the core. The first component is assumed to be constant for conservative reasons, while the second component is time dependent and calculated from ANS standards; the time evolution of the core decay power is then represented by an exponential-like curve starting at 11691 kW and asymptotically decreasing to 4657 kW. Air inlet temperature of 30°C is assumed for the cooling tower, which is relatively high but conservative.

The simulation starts when the SBO event occurs: the pumps are tripped and the natural circulation in the pool is activated. The pool temperature, which in nominal conditions is between 50 and 60°C, rises to 90°C due to the insertion of two more batches coming from the core and the decrease of mass flow rate, switching from forced flow regime to natural circulation. The maximum temperature is reached in less than a day; after one day the natural circulation is established and the heat removed balances the decay heat, reducing the pool temperature. A similar time evolution is computed for the intermediate loop, for which the maximum and minimum temperatures are 74°C and 58°C, respectively. The air side shows a similar behavior and the air flow rate remains higher than in the nominal operating conditions for several days after the initiation of the transient.

This preliminary work demonstrates that a passive cooling system for the SFP may be effective in rejecting the decay heat to the air, used as ultimate heat sink, in case of SBO, and can prevent Fukushima-like events from damaging the spent fuel.

Further details are provided in the topical report "Spent Fuel Pool: Passive Decay Heat Removal System".

19. Spent Fuel Pool: Criticality Monitoring and Safeguards (Report I2S-FT-16-15)

Prepared by: Alireza Haghighat (Virginia Institute of Technology); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 35 pp.

This topical report documents the work performed in support of the design of the I²S-LWR spent fuel pool (SFP), including criticality safety and monitoring for safeguards analyses.

A number of spent fuel pool designs were considered and the design for the Westinghouse AP1000 was selected. The storage cell dimensions were modified to accommodate the I²S-LWR fuel assembly. Other pool parameters (e.g. inter-assembly spacing, and neutron absorber dimensions and boron concentrations) were selected by conducting a detailed safety analysis study to ensure pool subcriticality for normal and off-normal conditions. The event/accident scenarios were developed based on the NRC guidelines of 10 CFR Part 50. Analysis was performed to i) evaluate the required pool capacity, ii) decay heat generation and time to evaporation, iii) criticality safety analysis for both normal operation and accident scenarios, and iv) radiation worker biological dose equivalent.

SFP design is based on a modified AP1000 SFP design, accounting for the differences in fuel dimensions. SFP is assumed organized into arrays of 9x9 fuel assemblies, sized to accommodate 11 years of fuel plus a full core off-load. Metamic alloy sheets containing B_4C in-between assemblies providing additional reactivity control. For the conservative infinite lattice of such arrays containing all fresh fuel, SFP is subcritical with an ample margin (k=0.90490+/-0.00021).

Estimate of the decay heat is used to calculate the time to uncover fuel (i.e., for water level to decline to the top of fuel: depending on the assumptions it is estimated to be between 23 and 43 hours with no action and under very conservative assumptions. Access to additional makeup water is foreseen to guarantee at least 48 hours. However, as described in a companion SFP report, primary mechanism for reliably cooling SFP, even under SBO, is a passive decay heat removal system.

Several events and accidents scenarios were modeled, including change in pool water temperature or level, missing individual Metamic sheets, and distortion of lattice geometry under seismic events. In all cases the configuration remained subcritical, with a sole exception of unlikely change of geometry where the distance between all infinite lattice elements (fuel assemblies) would be reduced to less than half the original, with all fresh fuel.

Finally, the dose to workers (next/above the SFP) was assessed. With full water depth, the direct dose is negligible (10^{-9} rem/yr). The dose remains acceptable (below the regulatory limit) as long as there is at least about 3 m of water above the top of assemblies. Conservatively, the SFP is designed to have 4.4 m above the top of an assembly being transferred in/out of the pool.

In parallel to the standard design considerations, a novel tool was developed and computationally benchmarked that is capable of real-time simulation and monitoring of the SFP level of subcriticality. This tool is called RAPID (Real-time Analysis for Particle Transport and In-situ Detection) and is capable of calculating pool eigenvalue, subcritical multiplication factor, and pin-wise, axially-dependent fission densities, and detector responses. This level of detailed fission density information, which is not typically used in evaluation of used fuel facilities, is essential for real-time monitoring of criticality safety and material safeguards.

This report is divided into two sections: Section 1 presents the spent fuel pool design and safety analysis, and Section 2 presents details on the development of the RAPID tool.

Further details are provided in the topical report "Spent Fuel Pool: Criticality Monitoring and Safeguards".

20. Economics Analysis (Report I2S-FT-16-14)

Prepared by: Giovanni Maronati, Bojan Petrovic (Georgia Institute of Technology); Paolo Ferroni (Westinghouse Electric Company LLC); *Submitted by:* Bojan Petrovic (Project PI), Georgia Institute of Technology, Atlanta, GA (Dec. 2016). 58 pp.

Assessment of the construction cost of I^{2}S-LWR is an important aspect of the overall evaluation of the new reactor concept and its viability for commercialization, and it is presented in this topical report.

There are several approaches to cost estimation and economics evaluation of new nuclear power technologies. Frequently used guidelines rely on the Code of Accounts, originally developed in the U.S. Department of Energy (DOE) Energy Economics Data Base (EEDB) Program, proposed as evaluation tool by Hudson, and then published in the guidelines for economic evaluation of bids, by The International Atomic Energy Agency (IAEA). The code of accounts allows breaking down main costs (Total Capital Investment Cost, Fuel Cycle Cost, Operation and Maintenance) into individual systems and items.

In this study, a top-down differential economics evaluation approach was developed through the use of the Code of Accounts guidelines to assess the costs of the I^{2}S -LWR relative to a representative "mainstream" PWR. In this methodology, a representative PWR design was taken as a reference and the differential cost was estimated for each individual account based on the design difference (or similarity). Cost scaling techniques were applied to the accounts representing systems that differ to the ones of the reference PWR. Cost estimating techniques were used to evaluate cost of innovative components that are not part of standard PWR designs. In evaluating the cost difference of the I^{2}S-LWR from the standard PWR, the uncertainty in the estimate is reduced. A similar approach was used by ORNL to estimate the cost of a Fluoride-salt High-temperature Reactor (FHR).

A typical Westinghouse four-loop plant with a core thermal power of 3417 MWth was selected as the reference. Costs for that plant were prepared in 1978 by EEDB, averaging actual cost incurred in the construction of several nuclear power plants (NPP), itemized with a great level of detail according to the Code of Accounts. This best estimate costs are denoted PWR12-BE. For each account, the cost of equipment, site labor and site material is provided. The latest version of the account cost items was released in 1988, and later converted to January 2011 US dollars. Industry experts at Westinghouse Electric Company performed a "sanity check" of the cost items, adjusting the cost of several items to match the current supply chain data. Specifically, the equipment cost of account 222 (Main heat transfer transport system) was increased by $100 M, the equipment cost of account 227 (Reactor instrumentation and control) was increased by $75 M, and construction supervision on site cost was increased by $250 M. The resulting overnight cost of PWR12-BE was converted to 2016 USD, resulting in $4,090/kW. However, PWR12 has higher power output then I^{2}S-LWR. For a more appropriate direct comparison with I^{2}S-LWR, the cost of PWR12-BE was scaled (assuming traditional PWR technology) to I^{2}S-LWR power level (983 MWe) and denoted PWR10-BE. The overnight cost of PWR10-BE is $4,363/kW.

The detailed cost assessment of I²S-LWR was performed, systematically analyzing cost for each account, and applying the differential economics approach. First, Relative importance of each account, i.e., its contribution to the total cost was established, to help focus analysis on the most significant contributors. Moreover, the accounts describing components that are different than that of the PWR12-BE were then identified.

The main component contributing to direct cost is the main heat transfer system (Account 222). The system includes main coolant pumps, pressurizer and steam generation system (primary heat exchangers, intermediate piping). The steam generation system is different from that of a standard PWR as it made of innovative components (microchannel heat exchangers). The integral configuration has another implication on Account 221, which includes the Reactor Pressure Vessel (RPV), which is greater in size and has more control rods and internals. On the cost reduction side, the reactor coolant piping (in Account 222) is not present and the pressurizer is integrated in hemispherical head of the vessel. Safety systems are allocated in Account 223. The passive DHRS of the I²S-LWR consists of a helical coil intermediate heat exchanger placed in the RPV, a water intermediate loop, and a cooling tower with a water-to-air heat exchanger. A careful cost analysis of the items included in this account was performed. Turbine generator equipment (Accounts 23x) is not believed to be much different than that of the reference design. Factors were applied to scale the cost of these components to the power level of the I²S-LWR. I²S-LWR structures (Accounts 21x) mainly differ from that of a standard LWR in yardwork for the reactor containment vessel, which is partially below grade. The cost associated to this account will depend on the excavation depth that will be chosen.

Additionally, due to its compact Nuclear Island footprint, I²S-LWR facilitates (and includes in its reference version) the use of seismic isolators. The seismic protection relies on seismic isolators installed on the nuclear building sub-foundation, which are also included in these accounts. Two cases were considered, one with an assumed 0.3g peak ground acceleration (representing Central Eastern US), and the second one with an assumed 0.7g peak ground acceleration (representing Western US). The final comparison of the overnight costs in 2016 USD is summarized in the following table, which presents results for PWR12-BE and PWR10-BE, against the I²S-LWR for the Central Eastern US (0.3g seismic zone).

	PWR12-BE	*PWR10-BE (reference)*	*I²S-LWR (w/ isolators) 0.3g zone*	Reduction (from PWR10-BE)
Overnight Capital Cost ($)	4,678,993,592	4,290,079,900	4,159,619,495	130,460,405
Overnight Capital Cost ($/kW)	4,090	4,363	4,230	132 (3.04%)

The overnight capital cost (on the $/kW basis) of I²S-LWR is by 3.0% lower than that of PWR10-BE, but at the same time it is by 3.4% higher than that of the higher-power PWR12-BE. However, for the Western US (0.7g), benefits of the seismic isolation are more pronounced, and the I²S-LWR overnight capital cost is 10.4% lower than that of PWR10-BE, and by 4.5% lower than that of PWR12-BE.

Thus, the analysis performed in this study estimates that the advanced I²S-LWR design will enable cost reduction in the range of 3 to 10% as compared to a traditional design of the same power level (PWR10-BE). However, to put these results in the proper perspective, we emphasize that:

- Uncertainties in estimates of nuclear power plants have traditionally been very high, much higher than the differences estimated here.
- Use of the differential economics approach should significantly reduce the uncertainties, but the uncertainties in differences are likely still similar or larger than the estimated differences themselves.
- The analysis is based on the pre-conceptual design level of the I²S-LWR design.

In spite of all the caveats, it could be stated that the analysis indicates that I²S-LWR has potential to offer an economically attractive design, with overnight cost comparable to, and perhaps somewhat lower than that of a nuclear power plant based on a traditional PWR design. In other words, I²S-LWR design offers enhanced safety, without penalizing economics.

Further details are provided in the topical report “Economics”.

Part III – Path Forward

21. Path Forward

This section discusses possible path forward. Moreover, it identifies important outcomes of the I²S-LWR project, in particular those with potential broader impacts beyond this project.

21.1 Path forward considerations

The I²S-LWR project has successfully completed its objectives, accomplishing a number of technical achievements. However, equally (if not more) important is the question of possible path forward. The I²S-LWR Team has identified two high-level approaches (Figure 21-1) that promise significant long-term benefits.

- The self-evident path forward is to focus on the reactor concept itself, and to continue development of a new, evolutionary iPWR Gen-III++ design, i.e, to aim to move the I²S-LWR concept to a preliminary design, and ultimately to commercialization. Conceivably, this may become the next step toward the LWR sustainability.
- The other possibility is to focus on new technologies developed within the I²S-LWR project. While most of these technologies are relevant primarily for iPWRs and SMRs, some of them are also relevant for non-LWRs.

While at first glance these two approaches may seem exclusive of each other, they are in fact synergistic and support each other.

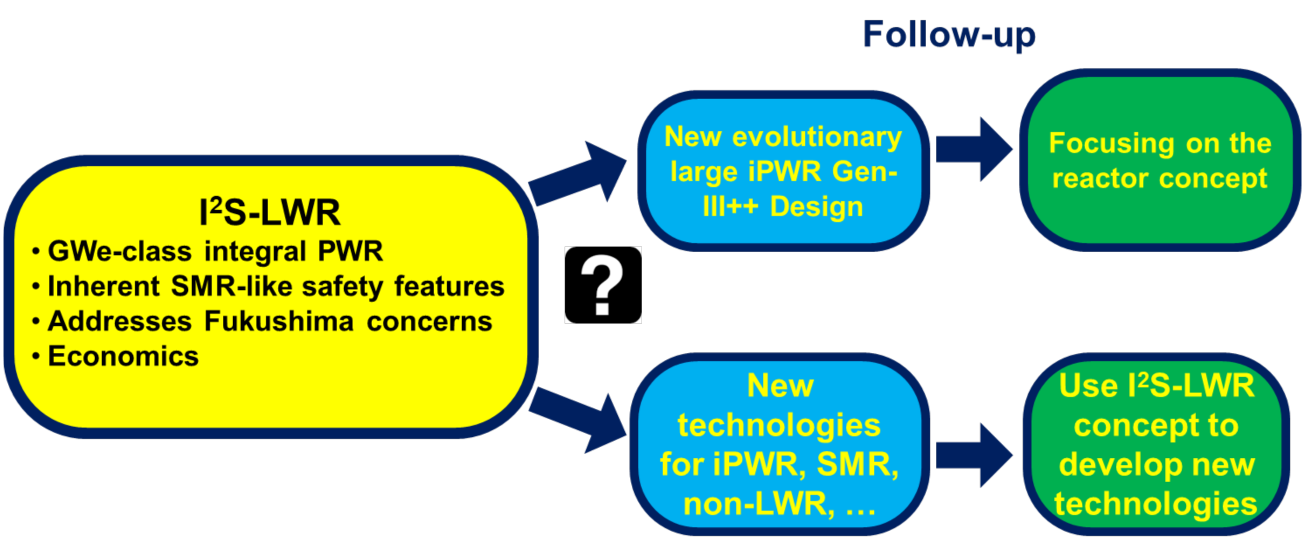


Figure 21-1: Potential Path Forward and Impact of I²S-LWR Project

21.2 I²S-LWR concept as a framework for development of new technologies

Development of new technologies by itself is a significant outcome irrespective of the ultimate deployment of I²S-LWR. Moreover, one can continue to use the I²S-LWR concept as a consistent framework for development of relevant new technologies. As illustrated in Figure 21-2, during this IRP project, the I²S-LWR concept benefited from adopting (and further developing) a range of technologies, including, for example, microchannel heat exchangers and integral primary circuit configuration. Moving forward, the I²S-LWR concept can serve as a consistent framework or testbed for continued development of these technologies, benefiting a range of reactor designs, and thereby in a sense "returning the investment" made during the IRP phase.

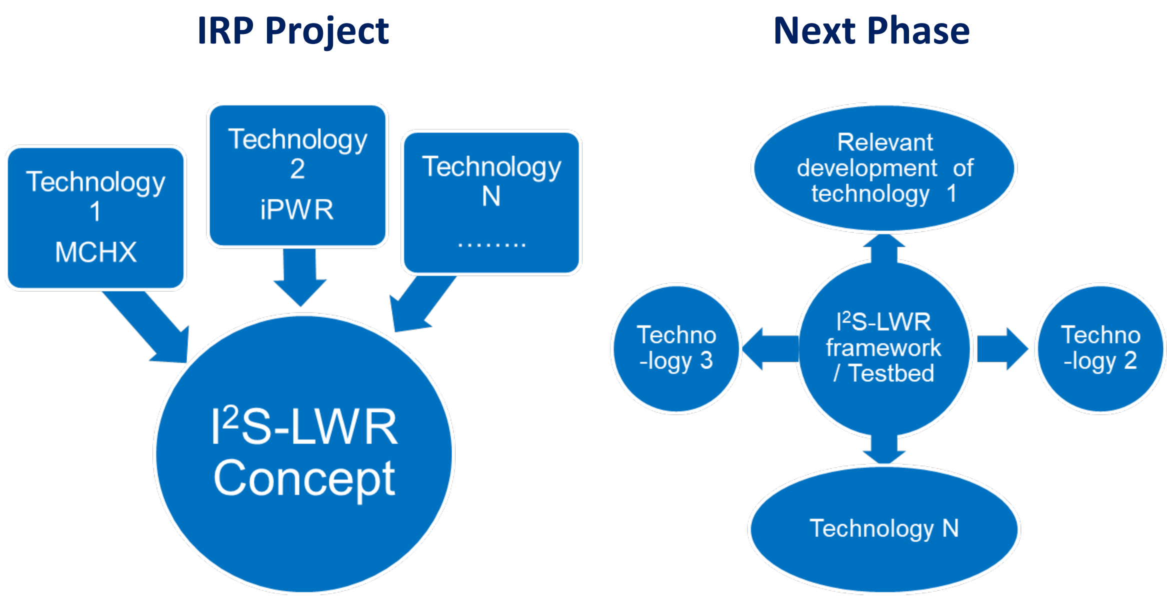


Figure 21-2: Benefits of using I²S-LWR project as a framework for development of new technologies

It is important to clarify the importance of having a realistic testbed. We will use the microchannel heat exchangers (MCHX) as an illustrative example. I²S-LWR uses integral vessel configuration and aims to achieve several times higher power output than what was previously considered feasible in iPWRs. This requires more compact components, including the core itself and the primary heat exchangers. Therefore, microchannel heat exchangers were selected. They provide significantly higher volumetric heat transfer capability, and a simplistic back-of-the-envelope estimate initially indicated that there should be no problem in designing them to fit within the available annular space (downcomer) within the I²S-LWR vessel. However, when various additional requirements and restrictions were considered coming from the detailed I²S-LWR vessel layout and from the performance requirements imposed by other primary and secondary loop components, it became clear that it will be quite challenging to satisfy them all simultaneously. For example, some of these requirements included:

- Assume the unit size not larger than what has already been proven manufacturable
- Add headers and connecting pipes to integrate multiple such units into modules, while maintaining near-uniform pressure drops
- Consider design to sustain full primary and secondary pressure
- Provide adequate space for installation
- Balance the MCHX primary side (core outlet temperature) and efficiency
- Harmonize the MCHX secondary side with the rest of the secondary loop, for efficiency
- Consider activation of MCHX, since they are located inside the vessel
- Consider dose to maintenance personnel, if a repair is necessary

These challenges were addressed and eventually resolved. However, most of these issues would not have surfaced in vacuum, or would not have had adequate "boundary conditions", were MCHX not integrated into a relatively well developed I²S-LWR concept. This importance of the testbed or framework applies not only to additional components, but also to approaches and analyses that have been developed in the I²S-LWR project.

21.3 Broad impact beyond the I²S-LWR project

Many outcomes (whether a component, technology, or analysis approach) of the I²S-LWR project potentially have a broader significance and impact beyond the project itself, and warrant considering further development. Main ones are listed below with a brief discussion.

1. <u>I²S-LWR concept.</u> As already mentioned, the I²S-LWR concept may be used as a framework/testbed for development of novel systems and components, in a consistent environment.
2. <u>Passive decay heat removal system (P-DHRS) for LWRs.</u> While P-DHRS was previously considered for high-temperature reactors, this project developed P-DHRS design for the significantly more challenging, due to lower temperature, LWR conditions. P-DHRS would resolve concerns of decay heat removal under SBO conditions (such as in Fukushima), since it provides indefinite cooling without external power. Further studies on optimum balance of economics and multiple/redundant P-DHRS would be useful.
3. <u>Passive spent fuel cooling.</u> Addresses SBO concerns (such as in Fukushima). Preliminary concept indicates viability, but further studies on practicality and economic impact are needed.
4. <u>Primary heat exchanger design based on micro-channel heat exchangers.</u> MCHX are of interest to LWR and non-LWR reactors, and in particular for reactor designs with an integral primary circuit configuration. Significant experimental and analytic work was performed, but more is needed to address all operational requirements, and to extend to other reactor types.
5. <u>Steam generation system (SGS) concept consisting of a single phase HX and flashing drums.</u> This SGS concept is of particular interest to reactor designs with an integral primary circuit configuration since it results in a more compact in-vessel portion of SGS, and thus enables smaller reactor vessel. However, the current design requires relatively large secondary pumps. It would be beneficial to study how to reduce their size and power consumption.

6. Fuel system with enhanced tolerance based on silicide fuel and FeCrAl or SiC cladding. While such fuel/clad system is also considered in the ATF program, it is there considered mainly for retrofitting current LWRs. In this project it was holistically incorporated into a new design, to maximize benefits and minimize drawbacks. This approach would be useful for advanced reactors designs incorporating ATF.
7. Theoretical basis for modelling of silicide (U_3Si_2) fuel swelling. This is important for all projects considering silicide fuel, since this is one of the potential show-stoppers. However, a number of important parameters that needs to be obtained from or supported by experimental irradiation measurements was not available when these models were developed. Future work should integrate into the models the irradiation results obtained after this project ended.
8. Materials (fuel/clad) properties database. This database comprehensively covers previous experimental results related to the materials of interest to this project. It is meant to serve as a reference database for consistent assumptions in design and analysis efforts. An update would be beneficial to the ATF community.
9. Materials testing. While it was limited in scope and budget, the experimental program produced relevant results supplementary to those obtained in other existing DOE programs. In particular, extending the fretting wear test would be useful.
10. Design of the integral reactor vessel and compact containment with safety systems. A fairly detailed layout and 3D CAD-CAM models of the I^{2}S-LWR vessel and containment were developed. While these layouts are specific to I^{2}S-LWR, many of their features may guide other integral designs.
11. High power density core design. Once the new experimental data on silicide fuel performance under irradiation are available, in particular on swelling, I^{2}S-LWR fuel analysis should be updated, and re-optimized to reflect the actual data. Additionally, the approach used to evaluate high power density core may be applied to other designs with similar objectives.
12. Advanced Cycle 1 core configuration. The developed approach to pseudo-equilibrium Cycle 1 configuration may be applied to other LWR designs to reduce the initial transitional fuel cycle penalty.
13. Differential capital cost evaluation. This approach offers a powerful tool to obtain a defendable cost estimate for reactor design in an early development stage.
14. Compact nuclear island design. Enables cost-effective implementation of seismic isolator, and is relevant for many reactor designs.
15. Ultrasonic primary flow meter for integral configuration. Proof of principle has been demonstrated. Further work to develop this technique would be very valuable to most if not all integral reactors.
16. SCORE system for Human Centered Design. SCORE system pilot was developed and applied to the I^{2}S-LWR vessel layout redesign activities to demonstrate potential benefits of using a human centered design. This approach could be extended to other components, systems, or different reactors.

21.4 Testing to support further development and licensing

I^{2}S-LWR concept is an integral PWR, and as such it shares many features with PWRs and with integral reactors in general. Its specific design also shares a number of features with SMRs. While it introduces notable modifications and enhancements to the traditional PWR design, it may still be regarded as an evolutionary step. Therefore, the required testing may be categorized as follows:

a) For the systems that are common with PWRs, there is generally no need for testing.
b) For the systems common with iPWRs/SMRs, I^{2}S-LWR may be able to use some of their testing.
c) Novel systems will require full testing.

For illustration, the second category may include:

- Out-of-core nuclear instrumentation, which is also developed and being tested (or planned to be tested) for several integral SMRs.
- Internal CRDMs (I-CRDM). They are considered for several iPWR/SMRs. Westinghouse performed initial testing of I-CRDM for Westinghouse SMR.
- Pressurizer integrated in the vessel head.

In each case the specifics will determine whether no further testing, some testing, or complete testing is needed. To illustrate using the previously listed examples:

- If a novel nuclear instrumentation is demonstrated (for an SMR) under a similar operating conditions (pressure, temperature, radiation field), this may be applicable to I^{2}S-LWR, with no testing or only minimal additional tests.
- Regarding the I-CRDMs, the critical question is the in-vessel performance of the magnetic drive assembly ("mag-jack"). It should be noted that I-CRDM in Westinghouse SMR and in I^{2}S-LWR are essentially of identical design and are subject to very similar environment (water pressure and temperature), and thus demonstrating its performance for the former should be sufficient for the latter. However, the active core height is different, so some limited additional tests related to the insertion time may be needed.
- Integral pressurizers are considered in several SMR designs, and most of them are conceptually similar. However, each one also has its own specific design features, and it is possible that full testing may be required.

The following components and systems will need to be considered in the testing program. Most can be tested at a reduced scale, but some will need to be tested at the full scale. Also, the level of the required testing will vary.

1. Silicide fuel performance under irradiation. (Some irradiation is already underway in ATR.) Special attention should be paid to swelling.
2. Fuel/clad system testing, progressively going from lead pin to lead assembly to lead batch.
3. Confirmatory experiments for DNBR correlation for the I^{2}S-LWR fuel 19x19 lattice may or may not be needed.
4. Internal CRDMs.
5. Integrated pressurizer.
6. Automatic depressurization system.

7. Microchannel heat exchangers (MCHX) modules.
8. Steam generation system (MCHX and flash drums with pumps, at prototypic conditions)
9. Passive decay heat removal system.
10. Containment passive safety systems.
11. Main coolant pumps.
12. Integral test to evaluate coupled performance under SB-LOCA of the reactor vessel and containment.
13. Nuclear safety instrumentation, specifically out-of-core in-vessel neutron detectors suitable to cover the full range of conditions, i.e., source, low power, and power range. (Being developed for SMRs.)
14. Flow meter for integral configuration. (Being developed for other integral PWRs.)

Type of tests to be performed include

- Basic engineering development tests
- Component separate effects tests
- Integral effects tests

Engineering development tests are primarily performed to demonstrate feasibility and verify engineering capability before proceeding to fabricate large scale or prototype components.

Component separate effects tests includes the design, fabrication, operation, and qualification of large scale prototype components. These tests may also provide data for verification of computer models, as well as help to establishes boundary conditions and equipment parameters for the integral effects tests.

Integral effects tests serve to evaluate integrated performance of multiple interconnected components and systems, and to provide data to validate thermal-hydraulic and system codes that are used for integral analysis of these systems. They should include adequate representation of all important structures, components and systems, interconnections and piping. The results are intended to demonstrate integrated performance, and elucidate interactions of all systems (safety and non-safety).

If the I²S-LWR project continues, one of the top priorities will be to expand these testing areas and requirements into a detailed and clear testing plan. The relatively complete and mature level of the I²S-LWR concept achieved in the current project will enable and facilitate this task.

22. Conclusions and Recommendations

The DOE NEUP IRP Project "Integral Inherently Safe Light Water Reactors (I^2S-LWR)" was performed during the period 2/2013-12/2016. The goal of the project has been to develop a concept of a 1 GWe PWR with integral configuration and inherent safety features, accounting for the lessons learned from the Fukushima accident, while keeping in mind the need for economic viability of the new concept.

The project was performed by a multi-disciplinary multi-organization team of 14 organizations, lead by Georgia Tech and including seven other US universities (Brigham Young University, Florida Institute of Technology, University of Idaho, University of Michigan, Morehouse College, University of Tennessee, and Virginia Tech), nuclear industry and utility (Westinghouse Electric Co. LLC and Southern Nuclear), national laboratory (Idaho National Laboratory), and three international academia partners (University of Cambridge, UK; Politecnico di Milano, Italy; and, University of Zagreb, Croatia). This diverse expert team ensured successful completion of the project, while the participation of industry provided a valuable practical expertise and sanity-check throughout the course of the project.

In addition to about 30 Co-PIs and senior team members, the project engaged 10 young faculty, researchers, scientists and post-docs, as well as close to 30 graduate (MS and PhD) students, and over 70 undergraduate students, most of them through senior design projects. Thus, more than a hundred young faculty/researchers and students were trained and had opportunity to work on a cutting-edge research, under realistic real-life R&D conditions. This education and training by itself provides an excellent "return on investment" to DOE.

The project was guided by the top level requirements that were established in the proposal, and updated and somewhat expanded during the early project phase. The requirements were formulated in terms of hard 'must satisfy' requirements, with additional soft 'it would be valuable to satisfy' stretch targets. All hard requirements have been met; in addition, some of the stretch targets have been met as well.

The final outcome of the project is a fairly well developed concept of an advanced, integral PWR at 1,000 MWe power level, with enhanced safety and estimated competitive economics.

A non-exhaustive list of the main accomplishments follows:

- Harmonized the overall concept
- Applied holistic view to integrate design features, safety and economics
- Compiled a materials (fuel/clad) properties database
- Developed a framework for human-centered design approach and demonstrated the concept on the vessel design
- Selected an advanced fuel/clad system (U_3Si_2 fuel with FeCrAl or SiC cladding) with enhanced accident tolerance
- Developed a silicide swelling model, important for assessing viability of silicide fuel
- Performed selected – limited but relevant – experiments related to fuel and cladding
- Envisioned a viable path to novel fuel deployment

- Established the baseline fuel assembly design (19x19) and core layout (121 FA)
- Established the shut-down and control rod banks requirements
- Developed an I²S-LWR core physics benchmark for cross-validation of core physics computational tools across the project
- Developed several options for the first core and equilibrium cycle and verified their acceptable performance
- Developed 2-batch and 3-batch refueling strategy with 18-month and 12-month refueling intervals
- Developed an advanced pseudo-equilibrium first core
- Performed fuel cycle cost analysis
- Evaluated Pu disposition capability of I²S-LWR [performed and funded by, and of special interest to the UK team partner organization, aligned with the UK research priorities]
- Developed a detailed layout of the integral reactor vessel, with all primary components and internals
- Developed, to the appropriate level of detail, information on main pumps, integrated pressurizer, internal CRDMs, core barrel and radial reflector, automatic depressurization system)
- Performed comparative flow induced vibrations (FIV) analysis
- Evaluated thermal performance of the high power density core
- Performed preliminary vessel stress analysis
- Selected the micro-channel type heat exchangers (MCHX) for the primary heat exchangers and performed and optimized module design,
- Built MCHX experimental testing facility and performed relevant experiments
- Introduced and evaluated novel steam generation system (SGS) concept, based on the in-vessel single phase primary heat exchangers (PHX) and out-of-vessel flashing drums (FD)
- Developed and optimized Power Conversion System (PCS) based on the SGS concept to achieve target efficiency
- Established a detailed functional scheme of PCS
- Assessed potential benefits of an alternative PCS based on the Kalina cycle
- Established I&C strategy and main operational control algorithms for the core and flashing drums; performed system modeling and simulations; examined stability and self-diagnostics
- Examined I&C aspects of ex-core/in-vessel nuclear instrumentation, including novel routing
- Established safety philosophy
- Developed the concept and design of a passive decay heat removal system (P-DHRS), with ambient air as the ultimate heat sink; optimized its design
- Developed functional requirements and sized other safety systems (passive reactor cavity cooling system, passive containment cooling system, pressure suppression system, accumulators or makeup tanks)
- Developed a functional scheme of the containment with safety systems
- Developed a physical containment layout, considering operational, refueling and maintenance requirements
- Identified and classified relevant transient and accident scenarios

- Analyzed or assessed relevant transient and accident scenarios (LOFC, MFLB, MCHX blockage, SBO, SB-LOCA, SPADS, RIA. MSLB, inadvertent DHRS actuation)
- Performed Level 1 PRA and used results to guide design modifications to improve CDF
- Estimated CDF of the optimized system; confirmed it met the target requirements
- Performed preliminary PRA uncertainty qualification
- Established used nuclear fuel (UNF) management approach
- Evaluated I²S-LWR fuel decay heat characteristics
- Developed the spent fuel pool concept with a P-DHRS, with ambient air as the ultimate heat sink; optimized its design
- Developed a fast simulation tool for enhanced safety and security monitoring of SFP
- Evaluated fast neutron fluence on reactor vessel to assess its lifetime
- Evaluated and optimized type and placement of the ex-core/in-vessel nuclear instrumentation detectors
- Evaluated activation of MCHX to inform maintenance activities
- Evaluated dose distribution inside and outside the containment vessel to inform radiation protection
- Evaluated gamma heating of the radial reflector and assessed cooling needs (cooling channels)
- Developed nuclear island layout
- Established the nuclear island seismic isolation concept
- Developed the NPP site layout
- Established the differential economics analysis framework
- Performed differential economics analysis and assessed economic competitiveness
- Identified potential path(s) forward
- Identified important outcomes of this project with potential broader positive impacts beyond this project
- Educated and trained over 150 students and young researchers
- Engaged senior-level students in 14 senior design projects related to I²S-LWR
- Documented results in reports (14 quarterly reports; over 3300 pages total) and over 60 peer-reviewed papers published in journals and conference proceedings

Further development of the I²S-LWR concept has a potential to lead to an attractive enhanced LWRs, perhaps the next wave of LWRs, bridging the gap to Gen IV.

Key areas and tasks needed to move the development forward include:

1. Critically review results of silicide swelling obtained by the completed and ongoing irradiations at ATR (within the DOE ATF program). If the silicide fuel is viable with respect to swelling, update swelling correlations. If swelling is too high, fall back to oxide fuel.
2. Critically review progress on FeCrAl and SiC cladding based on the experimental results obtained recently (within the DOE ATF program). Update options for advanced cladding (FeCrAl or SiC or both). Update properties based on recent measurements. Plan fretting wear tests for SiC.
3. Prepare testing plan. Prioritize.

4. Update (and redo analyses) core design based on the updated fuel/clad selection and properties. Update/revise fuel cycle cost.
5. Critically review the differential construction cost estimate. Identify which accounts require refined analysis to reduce uncertainty in the predicted cost.
6. Review experiments that may support validation of the passive decay heat removal system analysis. Most available ones are for high temperature reactors. Evaluate whether and what type of experiments are needed. (This will part of the testing plan.)
7. Perform PRA on multiple/redundant cooling towers for multiple passive safety systems to obtain optimum performance/cost configuration.
8. Refine design of all passive safety systems. (Some systems located in the containment are currently sized and their functional requirements are defined, but actual design is at the notional level.)
9. Refine and update, as appropriate, design of the vessel and primary circuit components.
10. Move forward the seismic isolation design (currently at notional level).
11. Evaluate manufacturing and cost of microchannel heat exchanger, possible improvements, and use of additive manufacturing techniques.
12. Carefully re-analyze and re-optimize steam generation system. The current configuration requires large secondary pumps, which take significant space and increase the footprint of the nuclear island. Consider a scaled test facility at several MW level.
13. Extend PRA to level-2, possible level 3.
14. Build case for eliminating off-site EPZ.
15. Prepare preliminary licensing approach/plan.

Moreover, many outcomes of the I²S-LWR project have a broader significance and impact beyond the project itself, and warrant considering their further development. For this development, the most technically sound and cost-effective manner would be to use the I²S-LWR concept as a framework/testbed for development of these novel systems and components, since it would provide a consistent environment with meaningful requirements and limitations. Potential development areas include:

1. Evaluation of I²S-LWR developments that are applicable to retrofitting current LWRs.
2. Passive decay heat removal system (P-DHRS) for LWRs.
3. Passive spent fuel cooling.
4. Primary heat exchanger design based on micro-channel heat exchangers.
5. Steam generation system (SGS) concept consisting of a single phase HX and flashing drums.
6. Fuel system with enhanced tolerance based on silicide fuel and FeCrAl or SiC cladding.
7. Theoretical basis for modelling of silicide (U_3Si_2) fuel swelling.
8. Advanced Cycle 1 core configuration (pseudo-equilibrium cycle 1).
9. Differential capital cost evaluation (applied to other novel reactor designs)
10. Compact nuclear island design and savings enabled by seismic isolation.
11. Out-of-core in-vessel nuclear instrumentation.
12. Ultrasonic primary flow meter for integral configuration.

Overall, the DOE NEUP IRP "I²S-LWR" has been a very successful project, having both a potential on its own to further evolve into an advanced LWRs, bridging the gap to Gen IV, or serving as a testbed for further developing individual technologies of broader impact that were conceived within this project, or supporting individual follow-up on these technologies. Equally important, the project has trained, educated and engaged over 150 researchers, including over 100 students and young researchers. Continuation of the project should be expected to provide an excellent "return on investment" to DOE through these both components.

Part IV – Appendices

Appendix A

Abbreviations and Acronyms

Team Members

GT, Gatech	Georgia Tech, Georgia Institute of Technology
UM	University of Michigan
INL	Idaho National Laboratory
PoliMi	Politecnico di Milano
VT	Virginia Tech
FIT	Florida Institute of Technology
WEC	Westinghouse Electric Company
UT or UTK	University of Tennessee Knoxville
BYU	Brigham Young University
UI	University of Idaho
SNOC	Southern Nuclear Operating Company

Other Abbreviations

ACC	Accumulators
ADS	Automatic depressurization system
ALWR	Advanced light water reactor
BOC	Beginning of cycle
CBC	Critical boron concentration
CMT	Core makeup tank
CV	Containment vessel
CVCS	Chemical and volume control system
DBA	Design basis accident
DD-DI	De-aerated doubly deionized (water)
DHRS	Decay heat removal system
DOE	Department of Energy
DSA	Dynamic strain aging
DVI	Direct vessel injection
EAB	External advisory board
EOC	End of cycle
ET	Event tree
FIV	Flow induced vibrations
FT	Fault tree
FY	Fiscal year
GA	Genetic algorithms
HCD	Human centered design
HM	Heavy metal
HPD	High power density (core)
HPT	High pressure turbine
HX	Heat exchanger
IFBA	Integral fuel burnable absorber
IPT	Intermediate pressure turbine
IRP	Integrated research project
I&C	Instrumentation and control
I^2S-LWR	Integral Inherently Safe Light Water Reactor
LEU	Low enriched uranium
LOCA	Loss of coolant accident
LOFC	Loss of forced circulation
LOFW	Loss of feedwater
LP	Loading pattern
LPT	Low pressure turbine
LWR	Light water reactor
MCHX	Micro-channel heat exchanger
MDP	Motor driven pumps
MFP	Main feed pump

MOC	Middle of cycle
MOGA	Multi-objective genetic algorithm
MOO	Multi-objective optimization
MOX	Mixed oxide (fuel)
MPET	Multi path event tree
MTC	Moderator temperature coefficient
MWd/tHM	Megawatt-days per ton (initial) heavy metal
MWd/tU	Megawatt-days per ton (initial) uranium
NEUP	Nuclear Engineering University Program
NPP	Nuclear power plant
NSPSO	Non-dominated sorting particle swarm optimization
NSSS	Nuclear steam supply system
PCCS	Passive containment cooling system
PCHE	Printed circuit heat exchanger (here same as MCHX)
PCS	Power conversion system
PDHRS	Passive decay heat removal system
PFD	Process flow diagram
PHE	Primary heat exchanger
PHPR	Passive high pressure recirculation
PLPR	Passive low pressure recirculation
PRA	Probabilistic risk assessment
PRCCS	Passive reactor cavity cooling system
PRZ	Pressurizer
PSA	Probabilistic safety analysis
PSO	Particle swarm optimization
PSS	Pressure suppression system
PWR	Pressurized water reactor
RCP	Reactor coolant pump
RMS	Root-mean square
RPV	Reactor pressure vessel
SBLOCA	Small break LOCA
SCA	Single channel (thermal-hydraulic) analysis
SFP	Spent fuel pool
SGS	Steam generating system
URD	Utilities requirements document
UTS	Ultimate tensile stress
TOX	Thorium oxide (fuel)
TRU	Trans-uranic
TWG	Technical working group
YS	Yield strength
ZDT	Zitzler-Deb-Thiele (functions)

Appendix B

Contributors to the project

The list aims to give credit to all individuals that contributed to the project during the whole project duration. It does not imply that all contributors were fully funded or devoted full time to the project.

Contributors to the project

NOTE: The list aims to give credit to all contributors to the project during the whole project duration. It does not imply that all contributors were fully funded or devoted full time to the project. Some contributors (some graduate students) were fully funded, some contributors were partially funded, while others were funded through a fellowship or a synergistic activity and thus contributed at no cost to the project. Similarly, the list does not aim to quantify the scope of individual contributions. It ranged from a short-term expert help, or a one-semester-long undergraduate research project, to multi-year fulltime graduate research.

Senior Personnel
Bojan Petrovic (Georgia Tech)
Farzad Rahnema (Georgia Tech)
Chaitanya Deo (Georgia Tech)
Srinivas Garimella (Georgia Tech)
Preet Singh (Georgia Tech)
Richard Neu (Georgia Tech)
Matthew Memmott (Brigham Young University)
Guy Boy (Florida Tech)
Indrajit Charit (University of Idaho)
Annalisa Manera (University of Michigan)
Thomas Downar (University of Michigan)
John Lee (University of Michigan)
Lycurgus Muldrow (Morehouse College)
Belle Upadhyaya (University of Tennessee)
J. Wesley Hines (University of Tennessee)
Ali Haghighat (Virginia Tech)
Paolo Ferroni (Westinghouse Electric Co.)
Fausto Franceschini (Westinghouse Electric Co.)
David Salazar (Westinghouse Electric Co.)
William Mack (Westinghouse Electric Co.)
Jason Young (Westinghouse Electric Co.)
Alex Harkness (Westinghouse Electric Co.)
Robert Ammerman (Westinghouse Electric Co.)
Matthew Smith (Westinghouse Electric Co.)
Jerzy Chrzanowski (Westinghouse Electric Co.)
Jay Schmidt (Westinghouse Electric Co.)
Bernd Laskewitz (Westinghouse Electric Co.)
Rick Wright (Westinghouse Electric Co.)
Nick Irvin (Southern Nuclear Operating Company)
Abderrafi M. Ougouag (Idaho National Laboratory)
George Griffith (Idaho National Laboratory)

Geoffrey Parks (University of Cambridge)
Marco Ricotti (Politecnico di Milano)
Nikola Čavlina (University of Zagreb)
Davor Grgić (University of Zagreb)
Dubravko Pevec (University of Zagreb)
Hans Garkisch (consultant)

Faculty/Research Faculty/Scientists/Engineers, Post-Docs
Dan Kotlyar (University of Cambridge)
Lucas Stephane (Florida Tech)
Kkochnim Oh (Georgia Tech)
Arnab Roy (Georgia Tech)
Ce Yi (Georgia Tech)
Dingkang Zhang (Georgia Tech)
Zhiyuan Liang (Georgia Tech)
Kevin Connolly (Georgia Tech)
Saam Yasseri (Georgia Tech)
Qian Lin (University of Michigan)
Victor Petrov (University of Michigan)
Andrew Ward (University of Michigan)
Mario Matijević (University of Zagreb)
Krešimir Trontl (University of Zagreb)
Lucas Stephane (Florida Tech)

Graduate Students (MS and PhD)
Sai Li (University of Cambridge)
Paul Wilding (Brigham Young University)
Ben Neff (Brigham Young University)
Gopal Jani (Florida Tech)
Yassine Rayad (Florida Tech)
Neha Suri (Florida Tech)
Pietro Avigni (Georgia Tech)
Christopher Chapman (Georgia Tech)
Michael Chin (Georgia Tech)
Timothy Flaspoehler (Georgia Tech)
Christopher Kingsbury (Georgia Tech)
Richard Hoffman (Georgia Tech)
Andrew Holcomb (Georgia Tech)
Ryan Hon (Georgia Tech)
L. Michael Huang (Georgia Tech)
Alex Huning (Georgia Tech)
Gabriel Kooreman (Georgia Tech)
Daniel Lago (Georgia Tech)

Daniel Kromer (Georgia Tech)
Matthew Lynch (Georgia Tech)
Matias Marquez (Georgia Tech)
Giovanni Maronati (Georgia Tech)
Alex Moore (Georgia Tech)
Kyle Ramey (Georgia Tech)
Thomas Winter (Georgia Tech)
Hubert Gibson (Georgia Tech)
Alessandro Banzatti (Politecnico di Milano)
Ankan Guria (University of Idaho)
Anumat Sittiho (University of Idaho)
Kalyan Chitrada (University of Idaho)
Anumat Sittiho (University of Idaho)
Somayeh Pasebani (University of Idaho)
Sultan Alsagabi (University of Idaho)
Michael Crabtree (University of Michigan)
Nicholas Holmes (University of Michigan)
Kellen McCarroll (University of Michigan)
Matt Neumann (University of Michigan)
Scott Richards (University of Michigan)
Mingjun Wang (University of Michigan)
Matthews Weeks (University of Michigan)
Eric Welch (University of Michigan)
Matthew Lish (University of Tennessee)
Nathan Roskoff (Virginia Tech)
Katherine Royston (Virginia Tech)
Vince Wang (Virginia Tech)
William Walters (Virginia Tech)
Valerio Mascolino (Virginia Tech)

Undergraduate Students (Research Projects)
Joseph Burns (Georgia Tech)
Byron Brantley (Georgia Tech)
Alexander Anton (Georgia Tech)
Charles Stephens (University of Tennessee)
Ryan Tarver (University of Tennessee)
Brooke McMurrer (University of Tennessee)
Maxwell Bowdon (University of Idaho)
Brandon Cisco (University of Idaho)
Jordan Argyle (Brigham Young University)
Jonathan Boyack (Brigham Young University)
Seth Pacheco (Brigham Young University)

Seth Babcock (Brigham Young University)
Jesse Carlson (Brigham Young University)
Trent Lai (Brigham Young University)
Nicholas Lewis (Brigham Young University)
Brittney Saenz (Brigham Young University)
Ben Neff (Brigham Young University)
Shane Gallagher (Brigham Young University)
Jack Webster (Brigham Young University)
Richard Fitzhugh (Brigham Young University)
Blake Branham (Brigham Young University)
Aaron Bush (Brigham Young University)
Austin Carter (Brigham Young University)
Stephen Hancock (Brigham Young University)
Jacob Ladd (Brigham Young University)
Trevin Lasley (Brigham Young University)
Yunju Lee (Brigham Young University)
Trevor Lillywhite (Brigham Young University)
Benjamin Olvera (Brigham Young University)
Connor Wood (Brigham Young University)
Michael Ensign (Brigham Young University)
James Richards (Brigham Young University)
Patrick Taylor (Brigham Young University)
Michael Malmberg (Brigham Young University)
Caleb Reidhead (Brigham Young University)
Landon Schofield (Brigham Young University)
Nicole Smith (Brigham Young University)
Brandon Loveless (Brigham Young University)
Connor Bartholomew (Brigham Young University)
Nathanael Neil (Brigham Young University)
Devin Kimball (Brigham Young University)

Undergraduate Students (Senior Design Projects)
Brian Barron (Georgia Tech)
Matthew Marchese (Georgia Tech)
Sterling Olson (Georgia Tech)
Paul Rose (Georgia Tech)
Michael Saunders (Georgia Tech)
Brian Schwartz (Georgia Tech)
Joseph Burns (Georgia Tech)
Lindsey Cooke (Georgia Tech)
Malu Mbungi (Georgia Tech)
Nicholas Piper (Georgia Tech)

William Thompson (Georgia Tech)
David Zwick (Georgia Tech)
Bruce Berry (Georgia Tech)
Timothy Collart (Georgia Tech)
Christopher Kingsbury (Georgia Tech)
Adrian Mitchell (Georgia Tech)
Kyle Remley (Georgia Tech)
Christopher Safouri (Georgia Tech)
Jack Bell (Georgia Tech)
Carlos Charry (Georgia Tech)
Phillip DiMascio (Georgia Tech)
Nash Dingman (Georgia Tech)
Brett Hanley (Georgia Tech)
Clint Powell (Georgia Tech)
Matthew Robinett (Georgia Tech)
Catherine Bartgis (Georgia Tech)
Christina Hamm (Georgia Tech)
Craig Martin (Georgia Tech)
Blake McManaway (Georgia Tech)
Shubhang Tandon (Georgia Tech)
Spencer Applegate (Georgia Tech)
Benjamin Bane (Georgia Tech)
Samuel Hammond (Georgia Tech)
Blane McManaway (Georgia Tech)
Jacob Panfel (Georgia Tech)
Puvithel Rajan (Georgia Tech)
Margaret Sudderth (Georgia Tech)
Sheree Tamaklo (Georgia Tech)
Austin Wallach (Georgia Tech)
David Zabriskie (Georgia Tech)
Kelsi Austin (Georgia Tech)
Richard King (Georgia Tech)
Sam Klein (Georgia Tech)
Tony Ly (Georgia Tech)
Devin McGowan (Georgia Tech)
Rebecca Cottrill (Georgia Tech)
Nicholas Jackson (Georgia Tech)
Wesley Luttrell (Georgia Tech)
Dylan Robideaux (Georgia Tech)
Jason Walker (Georgia Tech)
William Bryans (Georgia Tech)
Adnan Hashim (Georgia Tech)
Joshua McCann (Georgia Tech)

Ayuko Morikawa (Georgia Tech)
Syfuddin Rashid (Georgia Tech)
Varija Agarwal (Georgia Tech)
Baron Carleton (Georgia Tech)
Mitchell Carr (Georgia Tech)
Enoch McKie (Georgia Tech)
Hunter Smith (Georgia Tech)
Diego Carvallo (Georgia Tech)
Jonathan Gates (Georgia Tech)
Briana Grant (Georgia Tech)
Brian Lee (Georgia Tech)
Amelia Tee (Georgia Tech)
Hemin Noorani (Georgia Tech)
Shawn Tyler (University of Tennessee)
Jason Rizk (University of Tennessee)
Matthew Buttrey (University of Tennessee)
Kendall Minor (University of Tennessee)
Michael Cooper (University of Tennessee)
Moyd Alamri (University of Idaho)

Graduate Students (Advanced Design Projects)
Dennis Adams (Georgia Tech)
Theresa Cutler (Georgia Tech)
William Murrey (Georgia Tech)
Brian Sculac (Georgia Tech)
Josh Thomas (Georgia Tech)

Graduate Student Interns
Eric Yehl (University of California, Berkeley, Westinghouse Electric Co.)
Giovanni Maronati (Georgia Tech, Westinghouse Electric Co.)

Undergraduate Student Interns
Amanda Lindgren (Michigan State University, Westinghouse Electric Co.)
Jesse Carlson (Brigham Young University, Westinghouse Electric Co.)

Appendix C

Publications

DOE NEUP IRP "I2S-LWR": Publications

a) Books

1. G. A. Boy, *Tangible Interactive Systems: Grasping the Real World with Computers*. Springer, U.K. ISBN 978-3-319-30270-6 (2016). [I²S-LWR is taken as an example in a chapter of this textbook.]

b) Journal papers (published, accepted, submitted)

1. B. Petrovic, "The Integral Inherently Safe Light Water Reactor," *Nuclear Engineering International*, pp. 26-29 (March 2014). (**Invited**)
2. B. Upadhyaya, M. Lish, J. W. Hines and R. Tarver, "Instrumentation and Control Strategies for an Integral Pressurized Water Reactor," *Nuclear Engineering and Technology*, **47**, 148-156 (2015).
3. A Guria and I. Charit, "Observation of serrated flow in APMT™ Steel," *Materials Letters*, **160**, 55-57 (2015).
4. T. Flaspoehler, B. Petrovic, "Radiation Damage Assessment in the RPV of the Integral Inherently Safe Light Water Reactor (I²S-LWR)," *European Physics Journal Web of Conferences (Special Issue ISRD-15)*, Vol. ***106***, 03004 (2016).
5. G. A. Boy, "Human-Centered Design of Complex Systems: An Experience-Based Approach," *Design Science Journal,* Cambridge University Press, U.K. (2017).
6. Benjamin A. Lindley, Dan Kotlyar, Geoffrey T. Parks, John N. Lillington and Bojan Petrovic, "Reactor physics modelling of accident tolerant fuel for LWRs using ANSWERS codes," EPJ Nuclear Sci. Technol. **2**, 14 (2016).
7. Dan Kotlyar, Geoffrey T. Parks, Enhancing plutonium incineration in the thorium-based I2S-LWR design with loading pattern optimization, Annals of Nuclear Energy, Volume 96, October 2016, Pages 401-411, ISSN 0306-4549, http://dx.doi.org/10.1016/j.anucene.2016.06.030.
8. Dan Kotlyar, Geoffrey T. Parks, Eugene Shwageraus, Thorium-based plutonium incineration in the I2S-LWR, Annals of Nuclear Energy, Available online 27 August 2016, ISSN 0306-4549, http://dx.doi.org/10.1016/j.anucene.2016.08.017.
9. Matthew R. Lish, Belle R. Upadhyaya, J. Wesley Hines, Development of I2S-LWR instrumentation systems, *Annals of Nuclear Energy*, **100**, 23-30 (2017).
10. Ryan Hon, Gabriel Kooreman, Farzad Rahnema, Bojan Petrovic, "Stylized whole-core benchmark of the Integral Inherently Safe Light Water Reactor (I²S-LWR) concept," *Annals of Nuclear Energy*, **100**, 12-22 (2017).
11. Dingkang Zhang, Farzad Rahnema, "Monte Carlo solution convergence analysis via COMET in a stylized integral inherently safe LWR benchmark problem," *Annals of Nuclear Energy*, **100**, 3-11 (2017).
12. M. J Memmott, P. R. Wilding, B. Petrovic, "An Optimized Power Conversion System Concept of the Integral, Inherently-Safe Light Water Reactor", *Annals of Nuclear Energy*, **100**, 42-52, 2017.
13. M. J Memmott, A. Manera, J. Boyack, S. Pacheco, M. Wang, B. Petrovic, "The Primary Reactor Coolant System Concept of the Integral, Inherently-Safe Light Water Reactor", *Annals of Nuclear Energy*, **100**, 53-67, 2017.
14. G. A. Boy, G. Jani, A. Manera, M. Memmott, B. Petrovic, Y. Rayad, L. Stephane, N. Suri, "Improving collaborative work and project management in a nuclear power plant design team: A human-centered design approach", *Annals of Nuclear Energy,* **94**, 555-565, 2016.
15. M. J Memmott, A. Manera "The Use of Flashing Drums and Microchannel Heat Exchangers to Generate Steam in Large Integral Light Water Reactors", *Nuclear Technology*, **191**, 199-212, 2015.
16. Thomas Winter, Chaitanya Deo, Comparison of fission gas swelling models for amorphous U_3Si_2 and crystalline UO_2, *Annals of Nuclear Energy*, **100**, 31-41 (2017).

17. Thomas Winter, James Huggins, Richard Neu, Preet Singh, and Chaitanya S Deo, "Fretting Wear Behavior of APMT Steel at 350 degrees C for Reactor Fuel Cladding Application," *MRS Advances*, **1**(35), 2495-2500 (2016).
18. Thomas Winter, Richard Hoffman, and Chaitanya S Deo, "Grain Subdivision Fission Gas Swelling Model for UO_2," *MRS Advances*, **1**(35), 2465-2470 (2016)
19. Ankan Guria, Indrajit Charit, Tensile properties of accident-tolerant aluminum-bearing ferritic steels, Annals of Nuclear Energy, 100, 82-88 (2017).
20. Kellen R. McCarroll, John C. Lee, Annalisa Manera, Matthew J. Memmott, Paolo Ferroni, Preliminary risk assessment of the Integral Inherently-Safe Light Water Reactor, *Annals of Nuclear Energy*, **100**, 89-102 (2017).
21. Oh, Kkoch Nim; Eom, Kwang Sup; Liang, Zhiyuan; Singh, Preet M., "High Temperature Oxidation Behavior of APM and APMT under Dry Air/Steam Condition," *MRS Advances*, **1**(35), 2471-2476, 2016.
22. Kromer, D. A., Huning, A. J., Garimella, S. (2017). "I2S-LWR Microchannel Heat Exchanger Design and Experimental Validation". *Annals of Nuclear Energy*. (Under Review)
23. M. Wang, A. Manera, V. Petrov, M. J Memmott, S. Qiu, "Passive Decay Heat Removal System Design for the Integral, Inherently-Safe Light Water Reactor (I²S-LWR)", *Annals of Nuclear Energy*, (under review)
24. M. Wang, A. Manera, M. J Memmott, J. C. Lee, S. Qiu, "Preliminary Design of the I²S-LWR Containment System", *Annals of Nuclear Energy*, (under review)
25. Andrew Ward et al., Establishing a Neutronics Design and Equilibrium Cycle Analysis for the I2S-LWR Reactor with UO2 and U3Si2 Fuel. (under review)
26. Annalisa Manera et al., Preliminary Design of the I2S-LWR Containment System. (under review)
27. Annalisa Manera et al., Passive Decay Heat Removal System Design for the Integral Inherent Safety Light Water Reactor (I2S-LWR). (under review)
28. Ce Yi et al., Progress on the Single Channel Analysis and Flow Induced Vibration Assessment for the I2S-LWR Concept Design. (under review)
29. KkochNim Oh, Chaitanya S. Deo, Zhiyuan Liang, Paolo Ferroni, Bojan Petrovic, and Preet M. Singh, High Temperature Oxidation Behavior of APM and APMT under Dry Air/Steam Conditions, Submitted for Publication to ANE. (under review)
30. Andrew M. Ward, Mingjun Wang, Matthew D. Neumann, Matthew Memmott, Annalisa Manera, Thomas J. Downar, "A Simulation of I2S-LWR Selected Transients," *Annals of Nuclear Energy.* (under review)

c) Conference papers with peer review

1. A. Guria, I. Charit and B. Petrovic, Tensile deformation behavior of Al-rich ferritic steels for advanced light water reactors, *Proc. Frontiers in Materials Processing Applications, Research and Technology (FIMPART)*, June 12-15, 2015, Hyderabad, India (2015).
2. B. Petrovic, F. Franceschini, P. Ferroni, "Fuel Cycle Cost Trade-off Studies for I²S-LWR (Integral Inherently Safe LWR) Fuel Design Selection," *2013 LWR Fuel Performance Meeting (TopFuel)*, September 15-19, 2013. Charlotte, NC (2013).
3. B. Petrovic, "Integral Inherently Safe Light Water Reactor (I²S-LWR) Concept: Extending SMR Safety Features to Large Power Output," *Proc. 2014 Intl. Congress on Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, NC, April 6-9, 2014, Paper 14311, pp. 78-85 (2014).
4. M. Memmott, M. Marchese, B. Petrovic, "Integral Inherently Safe Light Water Reactor (I²S-LWR) Concept: Integral Vessel Layout," *Proc. 2014 Intl. Congress on Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, NC, April 6-9, 2014, Paper 14313, pp. 86-94 (2014).

5. P. Ferroni, B. Petrovic, G. Sjoden, M. Chin, "Preliminary Thermal-Hydraulic Feasibility Evaluation of the Integral Inherently Safe LWR (I²S-LWR) High Power Density Core," *Proc. 2014 Intl. Congress on Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, NC, April 6-9, 2014, Paper 14316, pp. 1753-1764 (2014).
6. M. Memmott, A. Manera, "The Use of a Flashing Drum to Generate Steam in the Integral Inherently Safe (I²S) Light Water Reactor," *Proc. 2014 Intl. Congress on Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, NC, April 6-9, 2014, Paper 14193, pp. 60-70 (2014).
7. B. Upadhyaya, M. Lish, J. W. Hines, "Instrumentation and Controls for an Integral Inherently Safe Light Water Reactor," *Proc. 2014 Intl. Congress on Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, NC, April 6-9, 2014, Paper 14280, pp. 71-77 (2014).
8. B. Petrovic, "Progress in Development of the I²S-LWR Concept," *Proc. 10th Intl. Conf. on Nuclear Option in Countries with Small and Medium Electricity Grids*, Zadar, Croatia, June 1-4, 2014. (Invited Presentation)
9. Manera, A., Memmott, M.J., "Design and trade-off of the Passive Decay Heat Removal System (DHRS) of the Integral, Inherently Safe Light Water Reactor (I²S-LWR)", Proceedings of the 10th International Conference on the Nuclear Option in Countries with Small and Medium Electricity Grids, Zadar, Croatia, June 1 – 4, 2014.
10. G. Sjoden, M. Chin, C. Yi, and B. Petrovic, "Assessment of Flow Induced Vibration Limits in Preliminary I²S-LWR Fuel Designs," ANS PHYSOR 2014, Kyoto, Japan, September 28 – October 3, 2014
11. D. Salazar, F. Franceschini, B. Petrovic, "I2S-LWR Equilibrium Cycle Core Analysis", *Proc. PHYSOR 2014*, Kyoto, Japan, September 28 – October 3, 2014.
12. K. Oh, R.T. Hoffman III, C. Deo, P. Ferroni, B. Petrovic, and P. M. Singh, "High Temperature Oxidation Behaviour of Al/Si-Containing Stainless Steels for Nuclear Fuel Cladding Applications," *Proc. 19th International Corrosion Congress (ICC)*, Jeju Island, Korea, November 2-6, 2014.
13. D. Salazar, F. Franceschini, P. Ferroni, B. Petrovic, "I2S-LWR Fuel Management Options for an 18-Month Cycle Length", *Proc. Advances in Nuclear Fuel Management V (ANFM 2015)*, Hilton Head Island, SC, March 29 – April 1, 2015.
14. D. Zhang, F. Rahnema, R. Hon, G. Kooreman, B. Petrovic, "COMET Whole Core Solution to an I2S-LWR Benchmark Problem," *Proc. Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear Applications (SNA) and the Monte Carlo (MC) Method*, Nashville, TN, April 19-23, 2015, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2015).
15. G. Kooreman, R. Hon, F. Rahnema, B. Petrovic, "Three-Dimensional Benchmark Specifications Based off of the Integral Inherently Safe Light Water Reactor (I²S-LWR) Concept," *Proc. Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear4 Applications (SNA) and the Monte Carlo (MC) Method*, Nashville, TN, April 19-23, 2015, on CD-ROM, American Nuclear
16. D. Kotlyar, G.T. Parks, E. Shwageraus, "Exploring the Feasibility of a Homogeneous Thorium-Plutonium Cycle for the I2S-LWR Design," Proc. of ICAPP 2015, Nice, France, May 3-6, 2015.
17. S. Li, D. Kotlyar, G.T. Parks, D.A. Salazar, F. Franceschini, B. Petrovic, "First core design for the I²S-LWR 18-month cycle," Proc. *PHYSOR 2016*, Sun Valley, Idaho, USA, 1–5 May, 2016 (accepted).
18. D. Kotlyar, G.T. Parks, "A multi-batch approach for enhanced plutonium incineration in the thorium-based I²S-LWR design," Proc. *PHYSOR 2016*, Sun Valley, Idaho, USA, 1–5 May, 2016.
19. G. Maronati, B. Petrovic, N. Cavlina, "I²S-LWR Top-Down Differential Economics Evaluation Approach," *Proc. 11th Intl. Conf. on Nuclear Option in Countries with Small and Medium Electricity Grids*, Zadar, Croatia, June 5-8, 2016.
20. R. Ječmenica, D. Grgić, M. Matijević, B. Petrovic, "Nuclear and Thermal Hydraulic Calculation of a Representative I²S-LWR First Core," *Proc. 11th Intl. Conf. on Nuclear Option in Countries with Small and Medium Electricity Grids*, Zadar, Croatia, June 5-8, 2016.

21. M.R. Lish, B.A. McMurrer, B.R. Upadhyaya, and J.W. Hines, "Primary Coolant Flow Measurement for Integral Pressurized Water Reactors Using Ultrasonic Technique," 2017 Topical Meeting on NPIC&HMIT, San Francisco, June 2017
22. M.R. Lish, B.R. Upadhyaya, J.W. Hines, and R.A. Tarver, "Simulation, Monitoring and Diagnostics for Integral Inherently Safe LWR (I2S-LWR)," Proceedings of the 9th ANS International Topical Meeting on NPIC & HMIT, Charlotte, NC, February 2015.
23. P. R. Wilding, M. J Memmott, "Design of the I2S-LWR Steam Generation System using Multivariable Optimization Schemes", Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP) 2016, San Francisco, CA, USA, April 17-20, 2016.
24. M. J. Memmott, R. L. Fitzhugh, J. Richards, J. Schaumann, "Preliminary Design of a Thermal Storage System for use with a Current Light Water Reactor", Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP) 2016, San Francisco, CA, USA, April 17-20, 2016.
25. M. Wang, A. Manera, M. J Memmott, S. Qiu, "Passive Decay Heat Removal System Design for the Integral Inherently Safe Light Water Reactor (I2S-LWR)", 16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH- 16), Chicago, IL, USA, August 30 – September 4, 2015.
26. M. Wang, A. Manera, M. J Memmott, J. C. Lee, "Preliminary Design of the I2SLWR Containment System", 16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-16), Chicago, IL, USA, August 30 – September 4, 2015. American Nuclear Society, Vol. 3, p. 1835-1847p
27. N. Roskoff, A. Haghighat, and V. Mascolino, "Experimental and Computational Validation of RAPID," *Proceedings of ISRD 16th Annual Meeting*, Sante Fe, NM, May 7-12, 2017.
28. N. Roskoff, A. Haghighat, M. Millett, and J. Leidig, "Benchmarking of the RAPID Tool for a Subcritical Facility," *Proceedings of INMM 57th Annual Meeting*, Atlanta, GA, July 24-28, 2016,
29. V. Mascolino, A. Haghighat, and N. J. Roskoff, "Evaluation of RAPID for a UNF Cask Benchmark Problem," Proceedings of *ICRS13+RPSD2016*, Paris, France, Oct 3-6, 2016.
30. W. Walters, N. Roskoff, and A. Haghighat, "A Fission Matrix Approach to Calculate Pin-wise 3D Fission Density Distribution," *Proceeding of M&C 2015*, Nashville, Tennessee, April 19-23, 2015.
31. W. Walters, N. Roskoff, and A. Haghighat, "Use of the Fission Matrix Method for Solution of the Eigenvalue Problem in a Spent Fuel Pool" *Proceedings of PHYSOR 2014*, Kyoto, Japan, Sep. 28-Oct 3, 2014.

d) Extended Summary

1. B. Petrovic, "Integral Inherently Safe Light Water Reactor (I2S-LWR) – Concept Overview," *Trans. Am. Nucl. Soc.*, **109**, (2013).
2. E. Welch, A. Manera, M. Memmott, P. Ferroni, M. Wang, J.C. Lee, "Preliminary Safety Considerations for the Integral Inherently Safe Light Water Reactor (I²S-LWR)," *Trans. Am. Nucl. Soc.*, **111**, 961-964 (2014).
3. K. R. McCarroll, and J. C. Lee, "A Simplified Multi-Path Event Tree for the Integral Inherently-Safe Light Water Reactor (I2S-LWR)," *Trans. Am. Nucl. Soc.*, **115**, 826 (2016).
4. T. Flaspoehler, B. Petrovic, "Calculating Gamma Heating in the I²S-LWR Radial Steel Reflector," *Trans. Am. Nucl. Soc.*, **113**, 1626-1629 (2015).
5. K. M. Ramey, B. Petrovic, "Power Density Uncertainties in 2D Full Core Serpent Simulations," *Trans. Am. Nucl. Soc.*, **113**, 1223-1224 (2015).
6. K. M. Ramey, B. Petrovic, "2D SERPENT Models for I²S-LWR Radial Reflector Studies," *Trans. Am. Nucl. Soc.*, **113**, 1225-1227 (2015).

7. J. Richards, R. Fitzhugh, M. Memmott, "Preliminary Design of a Thermal Storage System for Use with a Current Light Water Reactor", Trans. American Nuclear Society Winter Meeting 2015, Washington DC, USA, November 8-12, 2015.
8. P. R. Wilding, M. J Memmott, "Optimization of Nuclear Steam Generation Systems via Multi-Parameter Sensitivity Analysis", Trans. of the American Nuclear Society Winter Meeting 2015, Washington DC, USA, November 8-12, 2015.
9. V. Mascolino, A. Haghighat, and N. J. Roskoff, "Evaluation of RAPID for a UNF Cask Benchmark Problem," *Transactions of ANS Annual Meeting*, San Francisco, CA, June 11-15, 2017, (invited)
10. N. Roskoff, A. Haghighat, and V. Mascolino, "Analysis of RAPID Accuracy for a Spent Fuel Pool with Variable Burnups and Cooling Times," *Proceedings of Advances in Nuclear Nonproliferation Technology and Policy Conference*, Santa Fe, NM, September 25-30, 2016,
11. V. Mascolino, A. Haghighat, and N. Roskoff, "External SNF Cask Dose Calculation Using RAPID," *Proceedings of Advances in Nuclear Nonproliferation Technology and Policy Conference*, Santa Fe, NM, September 25-30, 2016.

e) Conference presentations without paper

1. A. GA. Guria and I. Charit, "Mechanical Properties and Serrated Flow in Al-Bearing, High-Cr Accident Tolerant Ferritic Steel, Mechanical and Creep Behavior of Advanced Materials, TMS 2017, San Diego, CA, Feb. 26-Mar. 2, 2017.
2. A. Guria and I. Charit, "Dynamic strain aging in accident-tolerant ferritic steels Creep, Deformation, Texture, Nano and Nuclear Materials IV (in honor of Prof. K.L. Murty)," Plasticity 2016, Jan. 2-9, 2016, Kona, Hawaii. (Conference presentation)
3. A. Guria and I. Charit, "Mechanical Properties of an Accident-Tolerant Ferritic Steel," Materials for Nuclear Applications and Extreme Environments, Materials Science & Technology 2015 Conference, Columbus, October 4-8, 2015.
4. I. Charit, M. Bowdon, S. Pasebani and S. Alsagabi, "Accident-Tolernat Fuel Cladding Materials for Advanced Light Water Reactors," SMD 2014 Technical Division Young Professional Poster, TMS Annual Meeting & Exhibition, San Diego, CA, Feb. 16-20, 2014.
5. D. Kromer, A. Huning, S. Garimella, "Microchannel Heat Exchangers for the Integral Inherently Safe (I²S) Light Water Reactor, *Technical Presentation* ASME 2015 International Mechanical Engineering Conference and Exposition – Houston, Texas
6. T. Winter, R. Hoffman, and C. Deo, Grain Subdivision Fission Gas Swelling Model for UO2, MRS Spring 2016, Symposium MD8: Multiscale Behavior of Materials in Extreme Environments, April 2016, Phoenix, AZ
7. R Hoffman III and C. Deo, Examination of High Burnup Structure using an "n-fold" Method Potts Model, 6th International Conference on Recrystallization and Grain Growth, July 2016, Pittsburgh, PA
8. T. Winter, C. Deo. Fission gas modeling in uranium silicide and uranium di-oxide, U.S. National Congress of Theoretical and Applied Mechanics 2014, East Lansing MI

f) PhD Dissertations

1. M.R. Lish, Development of Instrumentation and Control Systems for an Integral Large Scale Pressurized Water Reactor, PhD Dissertation, University of Tennessee, Knoxville, August 2016.
2. Richard T. Hoffman, "Kinetic Monte Carlo Simulations of Defect Evolution in Materials under Irradiation by Energetic Particles," Ph.D. Dissertation in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (December 2016).

3. Daniel Kromer, Ph.D. Dissertation, Georgia Institute of Technology, Atlanta, GA (expected completion in 2017)
4. Thomas C. Winter, "Investigation of Accident Tolerant Fuel & Cladding Options for Light Water Reactors," Ph.D. Dissertation in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (expected completion in 2017)
5. Nathan Roskoff (expected completion in 2018)

g) MS Theses (including Laurea)

1. Matias G. Marquez, "Silicide Fuel Swelling Behavior and Its Performance In I^{2}S-LWR," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (Fall 2015).
2. Kyle Ramey, "Development of Serpent-Based Methods for I2S-LWR Depletion Modeling and Sensitivity Studies," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (Dec 2016).
3. Ankan Guria, "Mechanical Behavior of Aluminum-Bearing Ferritic Alloys for Accident-Tolerant Fuel Cladding Applications," M.S. Thesis in Materials Science & Engineering, University of Idaho (Fall 2015).
4. Joseph R. Burns, "Reactivity Control of a PWR 19x19 Uranium Silicide Fuel Assembly," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (Fall 2015).
5. Matthew E. Lynch, "Assessment of Uncertainty in Decay Heat for the Integral Inherently Safe Light Water Reactor," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (Fall 2015).
6. Giovanni Maronati, "Optimization of passive decay heat removal systems for the integral inherently safe light water reactor (I2S-LWR)", M.S. Thesis (Laurea Magistrale) in Nuclear Engineering, Politecnico di Milano, 2014.
7. Alessandro Banzatti, "Preliminary study of the I2S-LWR spent fuel pool passive cooling system," M.S. Thesis (Laurea Magistrale) in Nuclear Engineering, Politecnico di Milano, 2015.
8. Thomas C. Winter, "Comparison of fission gas swelling models for amorphous U_3Si_2 and crystalline UO_2," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (April 2016)."
9. Hubert C. Gibson, "Gaseous swelling and release in nuclear fuels during grain growth," M.S. Thesis in Nuclear Engineering, Georgia Institute of Technology, Atlanta, GA (June 2013).

h) Quarterly reports

14 quarterly reports with over 3,300 pages total

i) Senior and graduate design projects/reports

<u>GT Spring 2013</u>

1. Brian Barron, Matthew Marchese, Sterling Olson, Paul Rose, Michael Saunders, Brian Schwartz, "Developing vessel layout and the corresponding 3D CAD model," Design Team 3D, Georgia Tech Senior Design Project, Spring 2013.
2. Joseph Burns, Lindsey Cooke, Malu Mbungi, Nicholas Piper, William Thompson, David Zwick, "Fuel assembly design to enable compact core (with enrichment below 5%)," Design Team C1, Georgia Tech Senior Design Project, Spring 2013.

3. Bruce Berry, Timothy Collart, Christopher Kingsbury, Adrian Mitchell, Kyle Remley, and Christopher Safouri, "Fuel assembly design to enable compact core (with enrichment below 8%)," Design Team C2, Georgia Tech Senior Design Project, Spring 2013.
4. Jack Bell, Carlos Charry, Phillip DiMascio, Nash Dingman, Brett Hanley, Clint Powell, "Primary heat removal system based on internal micro-channel heat exchangers," Design Team X1, Georgia Tech Senior Design Project, Spring 2013.
5. Matthew Robinett, Catherine Bartgis, Christina Hamm, Craig Martin, Blake McManaway, Shubhang Tandon, "Primary heat removal system based on internal compact shroud-and-tube heat exchangers," Design Team X2, Georgia Tech Senior Design Project, Spring 2013.
6. Spencer Applegate, Benjamin Bane, Samuel Hammond, Blane McManaway, Jacob Panfel, Puvithel Rajan, Margaret Sudderth, Sheree Tamaklo, Austin Wallach, David ZabriskieScoping shielding studies," Design Team S, Georgia Tech Senior Design Project, Spring 2013.
7. Dennis Adams, Theresa Cutler, William Murrey, Brian Sculac, Josh Thomas, "Passive Decay Heat Removal System (DHRS) – initial scoping," Design Team 3D, Georgia Tech Advanced Design Project, Spring 2013.

GT Spring 2015

1. Kelsi Austin, Richard King, Sam Klein, Tony Ly, Devin McGowan, "Integral Inherently Safe Light Water Reactor Silicide Fuel Performance Evaluation," Design Team I1, Georgia Tech Senior Design Project, Spring 2015.
2. Rebecca Cottrill, Nicholas Jackson, Wesley Luttrell, Dylan Robideaux, Jason Walker, "I2S-LWR Burnable Absorber," Design Team I2, Georgia Tech Senior Design Project, Spring 2015.
3. William Bryans, Adnan Hashim, Joshua McCann, Ayuko Morikawa, Syfuddin Rashid, "Activation Concern for I2S-LWR In-vessel Primary Heat Exchangers," Design Team I3, Georgia Tech Senior Design Project, Spring 2015.
4. Varija Agarwal, Baron Carleton, Mitchell Carr, Enoch McKie, Hunter Smith, "I2S-LWR Containment Design," Design Team I4, Georgia Tech Senior Design Project, Spring 2015.

GT Spring 2017

1. Diego Carvallo, Jonathan Gates, Briana Grant, Brian Lee, Amelia Tee, Hemin Noorani, "3D Layout Model of Integral Inherently Safe Light Water Reactor," Georgia Tech Senior Design Project, Spring 2017.

UTK Spring 2016

1. Shawn Tyler, Jason Rizk, Matthew Buttrey, Kendall Minor, Michael Cooper (team lead), "Using Ultrasonic Flowmeters in Integral PWR Instrumentation," University of Tennessee Senior Design Project, Spring 2016.

U. Idaho, 2015

1. Moyd Alamri, "Evaluation of Accident Tolerant Fuel Cladding Materials for LWR," Capstone Senior Design Project, 2015.

j) Other

1. B.R. Upadhyaya, M. Cooper, J. Rizk, K, Minor, M. Buttrey, and S. Tyler, "Development of Reflection Transit Time Ultrasonic Flowmeter for Coolant Flow Measurement in Integral Pressurized Water Reactors," Spring 2016 Newsletter, ANS Human Factors, Instrumentation, and Control Division (HFICD), 2016.